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Linear Accelerator Fuel Regenerator and Producer
LAFR and LAFP

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Abstract

A linear accelerator fuel generator is proposed to enrich naturally occurring fertile U-238 or thorium 232 with fissile Pu-239 or U-233 for use in LWR power reactors. High energy proton beams in the range of 1 to 3 GeV energy are made to impinge on a centrally located dispersed liquid lead target producing spallation neutrons which are then absorbed by a surrounding assembly of fabricated LWR fuel elements. The accelerator-target design is reviewed and a typical fuel cycle system and economic analysis is presented. One 300 MW beam (300 ma-1 GeV) linear accelerator fuel regenerator can provide fuel for 3-1000 MW(e) LWR power reactors over its 30-year lifetime. There is a significant saving in natural uranium requirement which is a factor of 4.5 over the present LWR fuel requirement assuming the restraint of no fissile fuel recovery by reprocessing. A modest increase ($\sim$ 10%) in fuel cycle and power production cost is incurred over the present LWR fuel cycle cost. The linear accelerator fuel regenerator and producer assures a long-term supply of fuel for the LWR power economy even with the restraint of the non-proliferation policy of no reprocessing. It can also supply hot-denatured thorium U-233 fuel operating in a secured reprocessing fuel center.

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I. Need for Linear Accelerator Fuel Regenerators and Producers

The nuclear power industry today is mainly based on light water-cooled thermal burner reactors (LWR's) either of the pressurized water-type (PWR) or the boiling water-type (BWR). These reactors have proven performance in providing reliable and economical power and they actually supply almost 10% of the electric power generated in the US today. The LWR's use low enriched U-235 fuel (LEU). Enrichment is obtained in government-owned gaseous diffusion plants where natural uranium containing 0.7% U-235 is enriched to approximately 3% U-235 content for use in the LWR's. The enrichment concentration is chosen for safe, economical operation. It takes about 6 tons of natural uranium to produce 1 ton of LEU for use in the LWR power reactors. The depleted 5 tons of uranium contains about 0.2% U-235. The 1 ton of LEU is fabricated into fuel elements and burned in the LWR at a conversion ratio of approximately 0.6 (fissile material produced to fissile material burned) to a total burnup of 30,000 MWD/ton of the LEU. The limitations on burnup in the LWR is controlled by the nuclear reactivity in supporting a self-sustaining chain reaction assembly. The burnup limitation is not due to radiation damage to the cladding materials (zircaloy) of the fuel elements. The spent fuel element upon removal from the LWR contains roughly 2% fissile material, about half of which (1%) is Pu-239 and half (1%) is the remaining unburned U-235. Assuming no recovery of this fissile material for reuse by reprocessing, the above described nuclear fuel cycle requires a consumption of 6300 tons of natural uranium for each 1000 MW(e) LWR power reactor over a 30-year production lifetime for the reactor. This includes initial core inventory. In effect only 0.5% of the natural uranium is utilized to make power and the net burnup amounts to only 5000 MWD/ton of natural uranium. Approximately 30% less fuel is required if recovery of the Pu-239 is obtained from the spent fuel element by chemical reprocessing. The requirement would then decrease to about 4300 tons of natural uranium.

The natural uranium resource in the US has been estimated to be in the order of 3×10^6 tons. This is for uranium which can be reasonably recovered at a cost of less than \$100/pound of yellow cake (U_3O_8). This resource then can only support a maximum of 480,000 MW(e) of nuclear power. There are other estimates which fix the uranium reserves at only 1.5×10^6, therefore, only 240,000 MW(e) of nuclear power would be supportable. There are a number of conservative estimates which indicate that the US will need somewhere in the neighborhood of 1,000,000 MW(e) of power at the turn of the century (circa 2000) and that 400 reactors (1000 MW(e) each) will share this requirement with other power sources such as coal, oil, and solar. Based on these values, LWR's supplied by the present nuclear fuel cycle cannot be considered as a long-term solution to the US energy problem. In fact, utility executives today are quite concerned about whether to invest in another generation of LWR's. The fast breeder reactor (FBR) has thus been put forward as an absolute necessity in ensuring the long-term establishment of a nuclear fueled economy.

The fast breeder reactor has a conversion ratio greater than 1.0 so that it allows converting essentially all the naturally occurring U-238 to fissile material for generating power. With the FBR, the uranium resource can ultimately be extended 200 times the present value and essentially an unlimited energy source then becomes available. However, a number of drawbacks can be listed for FBR's.

1. The fissile material concentration being 10% or more in FBR's, is much higher than in LWR's. In fact LWR's must supply the initial Pu inventory for the FBR's.

2. A new technology must be adopted for FBR's to replace the present LWR's. The FBR's are either liquid metal (Na) or gas cooled (He) which implies new and higher unit capital investments and new safety regulations and precautions.

3. Reprocessing of fuel from the FBR is an absolute necessity.

The present US administration policy on nuclear power is to indefinitely postpone reprocessing as a means towards impeding the possible proliferation of nuclear weapons. This policy, at once, further limits the nuclear fuel resource for nuclear power generation and tends to eliminate fast breeder reactor fuel cycles. Studies have, therefore, been initiated in the US to investigate alternative nuclear fuel cycles which do not depend on nuclear fuel reprocessing. These are being implemented under the Nuclear Alternatives Systems Assessment Program (NASAP)

Another series of studies is being conducted by an international group under the title of International Fuel Cycle Evaluation Studies (INFCE). Some of the fuel cycles proposed to date as alternatives include the following.

1. Heavy water moderated reactors such as the CANDU reactors which utilize natural uranium fuel and obtain a net burnup of up as high as 10,000 MWD/ton of natural U. This is almost two times as much as that which can be obtained with the LEU-LWR fuel cycle. However, the heavy water reactor technology is not readily available in the US.

2. A number of variations of heavy water moderated reactors are suggested starting with light water and shifting to heavy water in a spectral shift reactor or a "hand me down cycle", burning the element first in an LWR to 2% fissile fuel and then in an HWR to less than 1%.

3. Switch to a thorium cycle in an LWR where the conversion ratio can be higher ($\sim$ 0.70). The thorium fuel resource is somewhat higher than the uranium resource in the US. However, in order to utilize thorium, a means of producing fissile U-233 would be necessary and this requires reprocessing in addition to converting and breeding.

4. Finally and most important, it is suggested that reprocessing for convertors and breeders be allowed in safeguarded nuclear fuel centers.

Although the studies are not yet complete, the apparent conclusion is that without reprocessing, the best burner and convertor fuel cycles can do is stretch the nuclear fuel resource by not more than a factor of two and new HWR technology would have to be introduced. Further stretching in nuclear fuel resources would require safeguarded nuclear fuel reprocessing centers.

Our proposal of the lienar accelerator fuel regenerator and fuel producer (LAFR and LAFP alternatively called "electronuclear fuel producer") introduces a whole new dimension to the LWR fuel cycle economy. Essentially the LAFR and LAFP produce fuel in-situ with an external source of neutrons. As such, the LAFR can stretch the nuclear fuel resource by a factor of 4.5 or more without reprocessing thus providing a longer term supply of fuel for the LWR. The LAFP with safeguarded reprocessing can stretch the fuel supply to the ultimate of 200 times the present just as the FBR does, however, with the notable exception that it maintains the LWR technology and economy in place. The LAFR and LAFP can also produce U-233 from thorium without and with reprocessing, respectively, thus opening up thorium as a nuclear fuel resource. Still another option is possible with the linear accelerator driven reactor (LADR), although as will be pointed out later, this option is less attractive to utilities because of the complexity of the need of a linear accelerator at each power station.

The option we believe most viable and attractive is the linear accelerator fuel regenerator (LAFR) because (1) it does not require reprocessing, (2) it stretches the nuclear fuel resource significantly, and (3) it insures a continuing LWR economy. The LAFR is thus not a power reactor or a breeder. It is a fuel generator and when applied in-situ it is an enricher in the same functional sense as a diffusion plant but without the disadvantage of severely depleting the natural uranium resource. One LAFR can support three or more LWR's and can be considered in the same light as toll enrichment. The fuel producer does not impose an extension of new technology on the utility operator. The fuel producer is independent of the utility. The fuel producer supplies the utility with conventional LWR fuel. More detailed examination of the LAFR and LAFP fuel cycles and economics are made in later sections of this report. The only other external neutron source competitor to the LAFR is the fusion-fission hybrid. In the long run fusion-fission because of the energy economy in producing a neutron could replace the accelerator as a fuel producer. However, the successful demonstration of the feasibility of a fusion reaction is still many years away. There are definite operational advantages to the LAFR compared to hybrids. The accelerator fuel producer requires only an extension of present day technology. In this sense the accelerator is unique and really has no near-term competitor. It appears to be the missing link in the LWR nuclear fuel cycle chain.

II. Accelerator Design

The ability to confidently utilize linear accelerator technology today stems from a long-term development effort in obtaining reliable linear accelerator machines for use in high energy physics basic research over the past 35 years. In a sense, we are capitalizing on a spin-off of a large developmental investment in basic research instrumentation. The general size and capacity of the LINAC we would need for fuel production is in the range of 1 to 3 GeV protons, and 100-300 ma beam current, or a machine with a capacity in the range of 100 to 500 MW(e) beam capacity. The general concensus among most accelerator designers[(2)] is that a continuous wave machine can be built today with a reasonable efficiency of 50%, line power input to beam power output. Some even estimate efficiencies as high as 70%. There is no machine operating today with the specific matching characteristics required. Up to now the machines built were research tools; there was no need for a production machine. However, the research machines have approached the required values in a pulsed operation and have become highly reliable and maintainable. Fig. 1 gives a table of parameters of existing relevant accelerators. It is interesting to note the LASL-LAMPF approaches the energy required and the CERN-ISR circulates many times more current safely (40 amps) than is needed for the LAFR. In addition, there is ongoing development work on improvements in RF power sources which gives more confidence that a highly efficient and reliable machine c n be constructed today.

III. Neutron Yield and Physics

Experimental yield of neutrons per incident proton on various finite metal targets as a function of the protou energy is given in Fig. 2. Further discussion on yields and energy spectra are given in reference (1). Although deuterium particles may yield 30% higher neutrons than protons, the increased cost and complexity of the accelerator for deuterium acceleration more than compensates this advantage; thus protons are preferred. Fig. 3 gives recent Monte Carlo calculations for neutron yields in infinite targets of Pb and UO_2. The effect of water in the target lattice is also shown. The significance of this data is that the yield of neutrons is expected to be at least 30 neutrons/1 GeV-proton. This does not, however, take into account the additional neutrons produced due to fast fission for neutrons energies $\leq$ 15 MeV. Additional estimates have been made for the fast fission neutron yield. The values are used to estimate fissile fuel production rate as will be shown later in Fig. 14. We understand a recent measurement on neutron yields has been made at Dubna in the Soviet Union by Vesilikov et al[7] with 660 MeV neutrons on a massive U target. The report indicates a neutron yield about 50% higher than values used herein. This would mean that the production rate would be 50% higher than assumed in this work which would make the economic result look even better than obtained herein.

IV. Target Design and Engineering

There appears to be a concensus that linear accelerator technology is available today for construction of an efficient and reliable continuous wave high current proton accelerator in the 1 GeV-300 ma-300 MW power level range. The technology has been developed over the last several decades at the expense of the high energy physics basic research programs in several national and international laboratories around the world. However, practically no effort to date has been expended on a suitable target assembly design for production of fissile fuel from natural fertile material.

The criteria for the design of a suitable target assembly is as follows:

1. The proton beam should be in the range of 1 to 3 GeV, because of (a) sufficient proton penetration in a heavy metal target, i.e., in solid lead, 1 GeV protons have a range of only 60 cm and (b) the yield of neutrons per proton increases linearly with proton beam energy, i.e., it is about 35 neutron/proton in Pb at 1 GeV, so that for a given production rate of neutrons converted to fissile fuel, i.e., in the order of 1 ton/year, the current should be in the order of 300 ma.

2. Since the beam must operate in a vacuum and the target assembly must be cooled under pressure, it is preferrable not to have to maintain a thin window through which the beam must penetrate from the vacuum to the higher pressure. A thin penetrating window is necessary to minimize loss of beam energy over a larger area.

3. Since considerable heat will be developed in the target assembly, it must be safely and economically cooled. The heat recovered should preferrably be of a high enough quality to convert to power.

4. The radiation damage to the fertile fuel material must be minimized through proper selection of cladding material.

Considering a number of alternate target design assemblies, we have come to the conclusion that the one shown in Figure 4 adequately meets a workable design. The assembly is designed to generate fuel in a PWR-type fuel assembly bundle. The bundles are placed in pressure tubes which surround a central slot through which jets of liquid Pb-Bi metal is passed down along the length of the calandria assembly of pressure tubes. The 1 GeV-300 ma proton beam spreads from the beam transport tubes entering the side of the calandria vessel with no interfering walls and interacts with the jets or columns of falling liquid lead. Neutrons are spalled and evaporated isotropically from the lead by the high energy protons and the high energy spallation and evaporation neutrons enter through the walls of the pressure tubes and are absorbed by the PWR-type fuel element rods forming fissile Pu-239 from U-238 in-situ. The purpose of the jets is to disperse the dense Pb-Bi target so as to allow a longer penetration path in the target for the purpose of distributing the neutron flux to a level which will be tolerable from a power density point of view. The Pb-Bi has a vapor pressure of 10^{-4} torr at the expected temperature of $300^{\circ}C$, thus maintaining an adequate low vacuum condition for the proton beam. The Pb-Bi is collected at the bottom of the vessel and cooled in a separate circuit. A plan view of the arrangement is shown in Fig. 5, indicating the vacuum vessel boundary and the pressure tube with the internal PWR fuel assemblies. In order to prevent loss of neutrons the outside area of the calandria vessel is covered with a neutron reflector (i.e. graphite) including top and bottom sections. The assembly should be long enough to accommodate a full length PWR bundle (8 ft long).

For cooling purposes, because it is necessary to provide a hard spectrum, steam is used, but since a high heat flux is also expected, wet steam or two-phase evaporative cooling is used. The calandria tubes are made of zircaloy as is the cladding of the PWR elements. Wet steam prevents corrosion of the zircaloy. Fig. 6 shows a typical PWR-type pressure tube assembly and cooling circuit. Since top loading of fuel elements is required, a shroud tube and channel surrounding the element must be provided for counterflow of the steam-water coolant. The steam pressure should operate at about 2000 psi and outlet temperature no higher than about $600^{\circ}F$. A cross section of the fuel assembly in the shroud tube is shown in Fig. 7.

Another HWR-type target assembly arrangement is shown in Fig. 8. This is of the horizontal pressure tube-type assembly similar in concept to the CANDU reactors. The Pb-Bi jets are in the center slot with the pressure tubes surrounding. The fuel elements are inserted horizontally through the pressure tubes. Shorter elements can be used in this arrangement. A plan cross section of the HWR-type design is shown in Fig. 9.

The advantage of this design is that there is no reentry or shroud tube needed and the flow of coolant is straight through from one end to the other. A disadvantage is that the tube sheet seal through reflector and shielding end tube sheets must be provided so that the tubes have the capability of expanding and contracting due to heating and cooling during startup and shutdown while maintaining vacuum.

Neutronic transport calculations using the neutron cross section libraries meshing with the transport code scheme shown in Fig. 10, was used to determine neutron yield, flux, burnup, and fissile fuel production. Monte Carlo calculations for neutrons yield per GeV proton impinging on Pb and UO_2 targets are given in Fig. 11. It can be seen that for an infinite target and neglecting high energy fission in UO_2 that the spallation and evaporation yields of neutrons per 1 GeV proton is about the same as a statistical average of about 35 neutrons per proton. By inserting the water for cooling the target, the neutron yield is lowered somewhat. The values of neutron yields for various volume ratio of moderator water to UO_2 fuel and water density are given in Fig. 12. The statistical average goes from 27.6 neutron per GeV proton for pressurized water-type configuration (volume ratio $H_2O/UO_2 = 2$ and $\rho H_2O = 0.7$) to 30.9 for steam cooled tight packed assembly (volume ratio $H_2O/UO_2 = 0.5$ and $\rho H_2O = 0.175$), which is not a large drop from the infinite metal medium calculations given in the previous figure. The general geometry is shown in Fig. 13 for flux distribution calculations. The calculation of initial yield (no fissile material in fuel) is given in Fig. 14 for various neutron yields (including fast fission reactions), fertile material (Th or U cycle) and coolant-type (H_2O or D_2O) and density. The initial yields for an accelerator producing 1 GeV protons of 300 ma current, vary from a low of 0.81 tons U-233/year for the Th-UO_2 system to 1.28 tons Pu-239/year for the UO_2-H_2O system. It is interesting that light water is a better coolant than heavy water because of the harder spectrum thus producing more fast fission neutrons. Furthermore, the thermal neutron yield for U-238 fission is higher than the Th-232, resulting in higher production rates for Pu-239. It should be pointed out that as fissile fuel builds into the system or when fuel containing fissile fuel is regenerated, the fissile yield in the highly undermoderated assembly should generate more fuel than when there is no fissile fuel content because of the multiplication of neutrons. For purposes of the systems and economic evaluation, we have assumed an average production rate of 1.2 metric tons/yr (1200 Kg/yr).

A parametric study was made of the thermal hydraulics of the system. The heat transfer characteristics of the assembly for the two phase coolant (steam-water) is shown in Fig. 15. The average heat flux of 197,000 BTU/ft^2/hr is equivalent to about 0.62 MW/m^2. For various steam inlet qualities ranging from 30% steam in water to 70% steam in water, the average steam concentration ranges from 80% steam to 92% steam for coolant velocities ranging from 25 to 75 ft/sec (and for various fractions of the heat flux), which is reasonable for evaporative cooling purposes while avoiding drying out the steam mixture. Wet steam is required to prevent the corrosion of the zircaloy metal cladding on the elements and the pressure tubes.

Fig. 16 indicates the pressure drop range for the two phase flow as a function of inlet steam quality for the same range of parameters given in Fig. 15, range from 5 to 40 psi which is entirely reasonable. The pumping power under the highest condition does not exceed 30 MW which is only 5% of the power required for the accelerator (600 MW) and under optimized conditions would be much less.

V. Fuel Cycle System

The conventional LWR nuclear fuel cycle is shown in Fig. 17. It requires 6300 tons of natural uranium for a 30-year lifetime 1000 MW(e) LWR with 75% power factor. Without reprocessing 1050 tons of spent fuel containing $\sim$ 2% U-235 + Pu-239 must be disposed of. With reprocessing and Pu recycle, the fuel requirement is reduced to $\sim$ 4300 tons of natural uranium. The proposed LAFR fuel cycle without reprocessing is shown in Fig. 18. The basis of this fuel cycle is first to (1) head-end enrich natural uranium to 1.6% U-235, then to (2) fabricate elements and generate fissile material in-situ in the LAFR to an average of 2.8% for the initial core inventory, (3) burn the fuel in an LWR for 30,000 MWD/ton down to 2% fissile material, (4) return fuel and regenerate in the LAFR back to 3.2%, (5) further burn another 30,000 MWD/ton in the LWR, and (6) finally discard the 2% spent fuel after the second LWR burn cycle. There are two reasons for head-end isotope enrichment: (1) it makes up inventory lost in the spent fuel, (2) it is more economical than building up initial core inventory from natural uranium in the LAFR, and (3) it takes less time to build up LWR inventory (i.e., it would take 1 LAFR at least two years to build up inventory of 2.4 tons for 1 LWR core loading). With 1.6% enrichment, it takes less than 1 year to provide the LWR inventory. The cycle assumes a 300 MW beam LAFR producing 1200 Kg/yr (1.2 MT/yr) of fissile Pu-239. Two burn cycles for a total of 60,000 MWD/ton is possible today with conventional zircaloy clad UO_2 fuel elements since burnup exceeding this value has been obtained in tests in existing water reactors. Zircaloy material damage saturates at 10,000 MWD/ton and further burnup does not alter the physical and chemical properties of zircaloy.(4) The 30-year lifetime natural U fuel requirement for this cycle is 1400 tons/day which is 4.5 times less than the 6300 tons/day presently required. This, therefore, yields a substantial improvement in utilization of the nuclear fuel resource. The calculation of fuel requirement is shown in Fig. 19. A shuffling of fuel between the LWR and LAFR is necessary in three zones. The average initial feed enrichment is 2.8%. Only one example set is given in the figure but the LAFR actually supplies three LWR's. In the equilibrium mode, fuel goes into the conventional LWR at 3.2%, progresses through the three zones after three years at 10,000 MWD/ton/yr and comes out at 2% for regeneration. The initial 1.6% enriched fuel which precedes the LAFR actually provides inventory for the LWR every 6 years because of the total burnup of 60,000 MWD/ton. The maximum equivalent burnup stress due to the regenerator is 6,000 MWD/ton which is not more than 20% of the LWR burn cycles. This is a maximum burnup based on generating no more heat in the target than is necessary to provide power to the accelerator to make the system self-sufficient. It may be more economical and less stressful not to generate that much power and to purchase outside power for running the accelerator. Optimization studies need to be formed on this point. Another interesting and attractive observation of this fuel cycle is that the isotope enrichment plant requirement to fuel LWR's is drastically reduced as the enrichment decreases. A factor of 6 is needed in natural fuel when enriching from 0.7% to 3.2%, while there is only a factor of 2.8 needed in enriching from 0.7% to 1.6%. As will be shown later, the capacity and separative work unit (SWU's) requirements are considerably reduced and thus the fuel cycle cost is reduced accordingly. Another point for this fuel cycle is that because of the 2 burn cycles there is only 500 tons of spent fuel for disposal or half the amount of the conventional LWR cycle. However, the spent fuel will contain mostly Pu-239 instead of half Pu-239 and half U-235 of the contained 2% fissile material.

A general calculation to indicate the effect of additional burn cycles is shown in Fig. 20. Beyond 2 burn cycles the incremental gain in resource decreases, i.e., 2 burn cycles improves the resource 4.5 times and 5 burnup cycles increase it further by a factor of 2.5 to 11.3 times. Although significant, the stress at 150,000 MWD/ton on the fuel element material may not be achievable. Another point is that even at 1 cycle, the gain is a factor of 2.3 which is significant even without stressing the element beyond today's conventional burnup.

Figure 21 traces through the entire U-Pu-239 fuel cycle economy for 1 LAFR supporting 3 LWR's. In this case, we have assumed a maximum thermal power generation in the target assembly of 1800 MW(t) (300 MW(t) direct beam deposition in the lead target and 1500 MW(t) in the surrounding blanket due to fast fission multiplication). This energy would be sufficient to generate 600 MW(e) at 33% power cycle efficiency to feed the accelerator power supply with a 50% power input to beam power output efficiency. The LAFR then becomes self-sufficient in power. Definitive design calculations are yet to be performed concerning the power cycle. Optimization and economic studies may indicate that it would be more desirable to generate less power in the target and purchase deficit power from an outside power source. The tradeoff between internal power generation and external purchasing of power, however, should not effect the economic comparison significantly.

VI. Economic Analysis

A first-order comparative economic analysis is presented in the following. Capital costs for the 1 LAFR/3-1000 MW(e) LWR's system is given in Figure 22. Since detailed estimates were not made, direct unit costs were used based on recent studies. The LINAC accelerator unit cost was assumed to be \$600/KW(e) of power input based on a detailed parametric study by P. Grand.[3] The target reactor assembly was assumed to be equivalent to an LWR at \$600/KW(e) based on LWR cost studies by Bechtel.[5] In some respects, the target assemly is less complex than an LWR. It is a subcritical assembly and has no control rods. On the other ahnd, it uses a liquid metal lead target and two-phase steam-water cooling. These were assumed as tradeoffs and the unit cost equivalent to an LWR is thus justified. The direct cost for the LAFR complex thus amounts to \$720 x 10^6. Adding the 3-1000 MW(e) LWR's brings the total investment for the system to \$2,520 x 10^6 or a unit capital cost of \$840/KW(e) of net power output. This value is 40% higher than the conventional \$600/KW(e) for LWR's and is the same range as first projected for the fast breeder reactor (FBR) capital cost. As will be shown later, this increased capital investment is traded off against lower fuel cost as in the case which is made for justifying FBR economics.

The power generation cost is shown in Fig. 23. The 1 LAFR-3 LWR cost is compared to the conventional LWR. There is an 83% increase over base capital cost for completion and financing for operation in 1986. The reason for this large increment is due to financing charges during the years of construction and escalation at 7% per annum, usually assumed.

The power generation cost is calculated averaged over the first 10 years of operation. As shown, capital charges (15% and 70% plant factor) are 40% higher for the LAFR system, however, fuel cycle cost is one third that of the LWR. An increment in operation and maintenance costs was assumed for the LAFR over the LWR due to the need to operate the LAFR. The total operating cost of 46.1 mills/kwh(e) for the LAFR-3 LWR system is thus only 6% higher than the 43.5 mills/kwh(e) estimated for the conventional LWR. Thus, the LAFR-3 LWR system is shown to be within reasonable competitive value with LWR costs today. The increased capital cost is traded off for decreased fuel cost, but much more significantly this is accomplished with a 4.5-fold increase in the nuclear fuel resource utilization.

A detailed breakdown of the fuel cycle is shown in Fig. 24. Escalation is assumed starting in 1977 to operation in 1986. Yellow cake and UF_6 conversion decreases 4.5 times because of the 4.5-fold decrease in natural uranium requirement and handling. The enrichment SWU requirements decreases first 4.5 times because of the lower throughput and another 2 times because of the lower enrichment (1.6% instead of 3.2%) which yield a total 9-fold decrement in enrichment cost. The remaining parts of the fuel cycle, i.e., fabrication,storage, carrying charges and transportation only declines 2 times because there is only a 2-fold decrease in fuel handling because of the 2-cycle burn. The overall 1 LAFR-3 LWR fuel cycle cost thus comes out to be about one third that of the conventional LWR fuel cycle cost.

It is interesting to point out that a recent Sargent and Lundy[6] estimate for LWR's operational in 1990 escalated and averaged for 30 year operation to the year 2020 estimates a total power generation cost of 80 mills/kwh(e) and a fuel cycle cost of 27 mills/kwh(e) which is about double the costs estimated above. The largest factor in the fuel cycle cost becomes the cost of yellow cake which reaches \$100/lb U_3O_8 in 1990 and escalates from there, reflecting the continued severe shortage of uranium reserves.

Another method of comparing fuel cycle cost is given in Fig. 25, estimating the cost of generating a gram of fissile Pu-239 directly in the LAFR alone. The \$720 x 10^6 capital investment in LAFR is excalated to \$1,320 x 10^6 for 1986 operation. Adding fuel cycle and O&M to the depreciation on the capital, brings the unit production cost to \$257.00/gm of fissile material produced (Pu-239). This also represents the unit cost of fissile material consumed in the LWR's and made up in the LAFR.

The equivalent cost of fabricated LEU-LWR fuel today for 1986 operation is \$242.00/gm of fissile material consumed (U-235 and Pu-239). At a conversion ratio of 0.6 the amount of power produced per gram of fissile fuel consumed is 18,200 kwh(e)/gm. Based on the actual cost of initially enriched U-235, the cost is calculated to be \$96.00/gm of U-235 contained in the element. However, when no fissile material is recovered because the spent fuel is thrown away and is not reprocessed, the amount actually burned is only 40% of that originally put into the reactor so that the effective charge is 2.5 times this amount for the material actually burned or \$242.00/gm. Again, on this basis the LAFR fuel cost is within 10% of the LWR cost and thus is reasonably competitive. A listing of the conclusions from the economic estimates is given in Fig. 26. By invoking the thorium and LWR cycles, the LAFR economics improves significantly.

Fig. 27 shows a conversion ratio correlation when going from fuel cycles of U-235-UO_2 to Pu-239-UO_2 to U-233-ThO_2 to U-233-ThO_2 (HWR moderated), the conversion ratio increases from 0.6 to 0.73 to 0.9 and the number of reactors that can be supported increases inversely proportional to one minus the conversion ratio (1-c.r.). Thus in Fig. 29, it is seen that for U-235-LWR, 3.4 LWR's can be supported; for U-233 and Pu-239-LWR, 5.0 LWR's can be supported and for U-233-ThO_2-HWR, 13.6 HWR's can be supported. The resource multiplication goes from 4.5 times to 6.7 times to 18.0 times without reprocessing. With reprocessing, all the resource can be utilized for an increase of 200 times that of the present fuel cycle resource utilization. The power production cost decreases from 44.3 to 39.7 to 35.9 mills/kwh(e). The latter does not decrease as much as one would expect because of the additional cost of heavy water (D_2O).

For a more proliferation-resistant thorium fuel cycle, it is suggested to denature the U-233 bred fissile material with the addition of natural or depleted U-238. In this manner, the U-233 is isotopically mixed with U-238 making it impossible to separate by chemical means when considering fuel reprocessing. It is suggested that about 20 to 30% of U-238 be mixed with Th-232, spiked with U-233 in a safeguarded fuel center which is then fabricated into LWR fuel. The LAFR and LAFP fits into this scheme very well in that, if natural uranium is enriched to approximately 7% in an enrichment plant as shown in Fig. 29 and is diluted with natural thorium 232 in a fuel fabrication plant with a proportion of 70% Th to 30% U, the fuel element produced will have about 3% U-235 ready for an LWR. The element after one cycle burnup can be regenerated in an LAFR to boost the fissile fuel content now in terms of U-233 back to 3% or more which can then be further burned up. On reaching the limits of regeneration after 2 or more cycles, the element can then be sent to a secured fuel reprocessing center to separate out the fission products and the fuel then refabricated and re-generated to LWR quality. In this manner, the fuel can be totally burned up. The cycle is doubly-proliferation resistant since it produces radioactively "hot" in addition to denatured fuel. The reason for the secure reprocessing area is that a small amount of Pu-239 (compared to present fuel cycle) is formed from the U-238 which is chemically separable. It should be pointed out here that if reprocessing is allowed, fissile Pu-239 can be readily produced from abundant sources of readily available depleted natural U-238 with the use of the LAFP.

An estimate of the effect of denatured systems on the US nuclear growth patterns has been made by the nuclear energy office of DOE and is graphically produced in Fig. 30. Essentially, it says that with a once-through LWR throwaway cycle and a fuel resource of \$3 x 10^6 tons of natural U that only 400-1000 MW(e) LWR's can be supported through the year 2010 before a U shortage developes. By recycling Pu and converting to thorium in a convertor and finally going to a Pu/U/Th FBR with secured reprocessing the LWR fuel can be increasingly extended but that not much more than one LWR reactor can be fueled by one FBR. This is symptomatic of FBR's since they essentially replace themselves with 15 to 30 year doubling times. This points up the adage that "fast breeders do not breed fast".

Performing the same analysis but now introducing the LAFR and the LAFB into the fuel cycle, as shown in Fig. 31, the 1 LAFR-3 LWR system can extend the fuel supply to support 700-1000 MW(e) reactors at the growth rate expected to the year 2030 and this is based on the more conservative estimate of 1.5×10^6 MT of uranium rather than the 3.0×10^6 MT assumed by the DOE in Fig. 30. With the 1 LAFR-5 LWR denatured thorium cycle and finally the LAFP with reprocessing, the utilization can be stretched into the long term future similar to the FBR with fuel reprocessing.

A comparison is now made between conventional, internal neutron (LWR and LMFBR) and non-conventional external neutron (hybrid and accelerator LAFR and LAFP) sources as shown in Fig. 32. A billion dollar U-235 enrichment plant buys about 1.5×10^6 SWU's and can support about 10-15 LWR's requiring 40,000 to 60,000 MT of natural fertile fuel depending on whether reprocessing is used or not. There is little flexibility in this fuel cycle. The uranium is depleted and the resource is severely limited. A 1000 MW(e)-LWR costs \$1 billion and roughly uses 6000 tons of uranium without reprocessing and 4000 tons with reprocessing as stated earlier. The net addition of U-235 or Pu-239 to make up for burnup is about 350 Kg/yr of fissile material. The LWR is severely limited in resource; it cannot support any other LWR. The LMFBR costs \$1.4 billion and depending on doubling time it can produce enough excess fuel to just about support one LWR. Without reprocessing, the LMFBR cannot function at all. Thus, the LMFBR is limited for useful support of any satellite LWR's.

The fusion-fission hybrid is estimated to cost $\$3.7 \times 10^9$ billion and produces 1.5 MT of fissile material while developing 500 MW(e) of power. Using the slightly enriched (1.6% U-235) fuel cycle discussed above, the hybrid can support five LWR's at a reduced resource requirement of 2100 tons of uranium over the 30-year lifetime while with reprocessing it essentially can burn up all the fuel. On a 0.9 conversion ratio thorium cycle, the hybrid can support 20 HWR's and there are essentially no limitations on the fuel cycle. With the accelerator, LAFR and LAFP, an investment of $1.3 billion produces 1.0 MT of fissile fuel which supports three LWR's or on the Th-U cycle 12 LWR's with a much reduced fuel resource requirement and an unlimited flexibility in fuel cycle. The essential differences between the hybrid and the accelerator is that the hybrid capital investment is estimated to be twice as great as the accelerator for the same fissile fuel production capacity and secondly, the hybrid needs feasibility demonstration, while the accelerator is here and can be reliably built today with only an extension of present day technology.

Still another comparison of long-term fissile fuel production systems is made in Fig. 33. In this figure, the FBR/LWR system is compared to the LAFR/LWR and the hybrid/LWR system in terms of total system capital investment and new technology investment (NTI) as well as R&D cost and commercial date of introduction. A constant total of 500-1000 MW(e) nuclear power economy is assumed for this comparison. The FBR/LWR requires a $600 billion investment with more than half, $350 billion, in new technology capital investment in LMFBR's. Since a good deal of R&D has already gone into LMFBR, it is estimated that only an additional $3 billion would be required to bring it to commercialization by about 1990 or somewhat thereafter. The LAFR/LWR system would require $630 to $725 billion total capital with $130 to $225 billion investment in the new technology of accelerator fuel regenerators and producers, depending on whether a U/Pu or a Th/U cycle is used. Since much effort has already gone into development of accelerators that has been paid for by the high energy physics basic research program, it is estimated that only another $3 billion would be required to bring it to commercialization, only a few years later than the LMFBR, say about 1995 or soon thereafter. In the hybrid/LWR system, the total investment cost ranges from $670 to $780 billion with new fusion-fission technology investment of $200 to $330 billion and would require much more R&D to bring to commercialization, say $15 billion and then it is highly questionable how long the development would take to become available after the year 2000, if ever. Undoubtedly from all viewpoints, i.e. technology availability, lower cost of new technology extension of fuel resource utilization, the LAFR/LWR system beats the FBR and the hybrid.

VII. Conclusions

Fig. 34 summarizes the unique features and advantages of the linear accelerator fuel regenerator and fuel producer. The last item has not beem mentioned heretofore but can be of utmost importance. Just as accelerators can transmute fertile material into fissile material through neutron reactions, so it should be possible to transmute very long-lived residual fissile material and transuranics into shorter lived and stable elements. It may also be possible to transmute long-lived fission products (i.e., 30-year half-life Cs-137 and Sr-90 to shorter lived and stable isotopes. This is a subject outside the scope of this paper but is important to investigate.

Finally, Fig. 35 estimates the schedule and R&D costs required for commercial implementation in terms of 1978 dollars. The program starts with neutron yield and kinetics studies probably at existing accelerator site (i.e., Los Alamos, Meson factor (LAMFP)) and progresses through accelerator and target development to a small-scale pilot plant, to a full-scale prototype and finally, near the turn of the century, to commercial plant operation. The entire research and development cost should total in the range of 3 to 4 billion dollars.

The accelerator fuel generator or electronuclear breeding can be considered to be the missing link in the thermal LWR and HWR nuclear power reactor economy and is eminently worthwhile pursuing to its full-scale development.

References

1. M. Steinberg, et al. Linear Accelerator-Breeder (LAB) A Preliminary Analysis and Proposal, BNL 50592 (November 10, 1976).

2. H. J. C. Kouts and M. Steinberg, "Proceedings of an Information Meeting on Accelerator Breeding", ERDA/Conf. 770107 (January 18-19, 1977).

3. P. Grand, BNL, private communication (April 1977).

4. R. Saybolt, Westinghouse Corp., private communication (January 1978).

5. Bechtel Corp., ANS Industry Report, 1976-7 (February 1977).

6. Sargent and Lundy, private communication (March 1978).

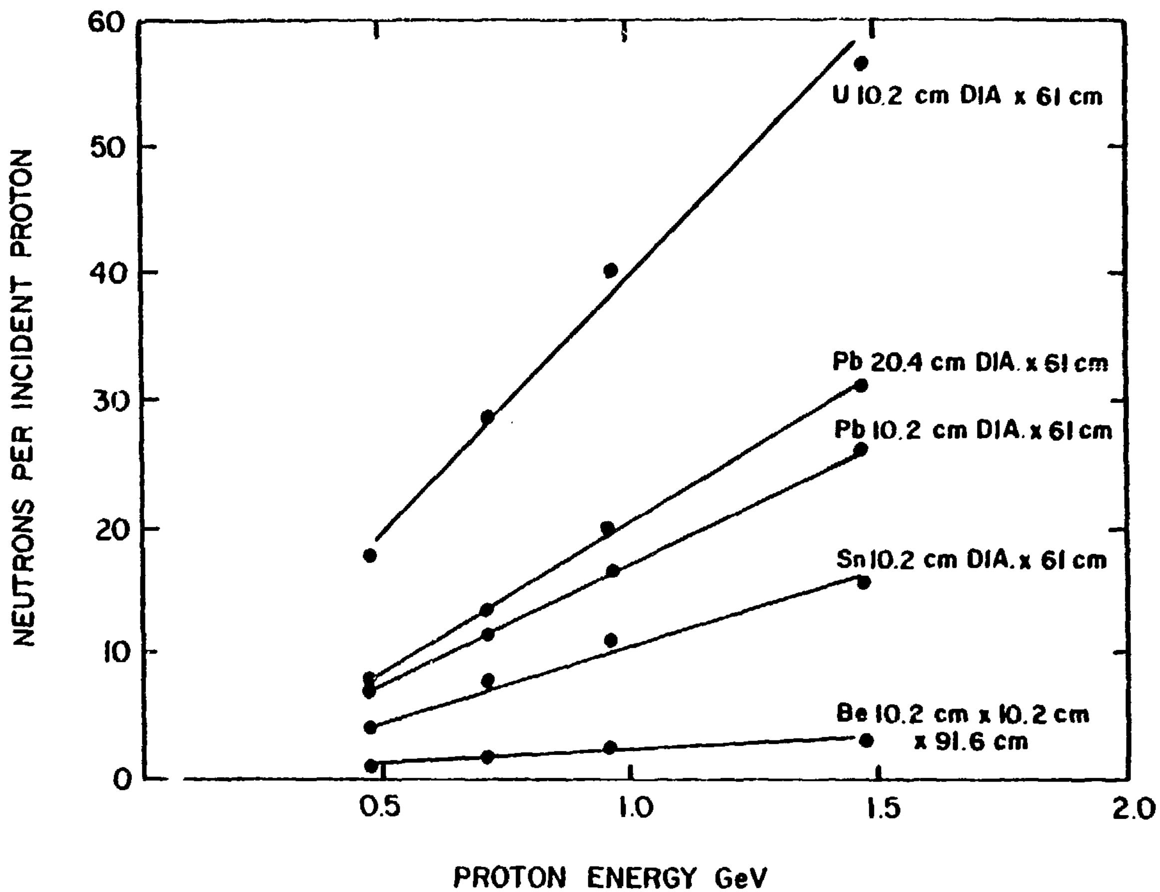
7. S. Schriber, private communication at meeting on Emerging Concepts in Advanced Nuclear Systems, Graz, Austria (March 29-31, 1978).

FIGURE 1

PARAMETERS OF EXISTING RELEVANT ACCELERATORS

LABORATORY	ACCELERATOR TYPE	ION TYPE	EMAX MEV	IMAX AMPS.	DUTY FACTOR	STATUS
LASL-LAMPF	LINAC	PROTON	800	0.020	12%	OP
LBL-HILAC	LINAC	H.IONS	10/NUCL.	-	50%	OP
BNL-AGS	LINAC	PROTON	200	0.2	0.5%	OP
FNAL	LINAC	PROTON	200	0.3	0.2%	OP
HEDL-HFNS	LINAC	DEUT.	35	0.1	100.0%	PROP
CERN-ISR	STORAGE RINGS	PROTON	$3X10^4$	40.0	-	OP
BNL-ISA	STORAGE ACCELERATOR	PROTON	$4X10^5$	6.0	-	PROP

FIGURE 2



EXPERIMENT YIELD OF NEUTRONS BY BOMBARDMENT OF A HEAVY METAL TARGET WITH HIGH ENERGY PROTONS

FIGURE 3

MONTE CARLO CALCULATIONS
FOR PB AND UO_2

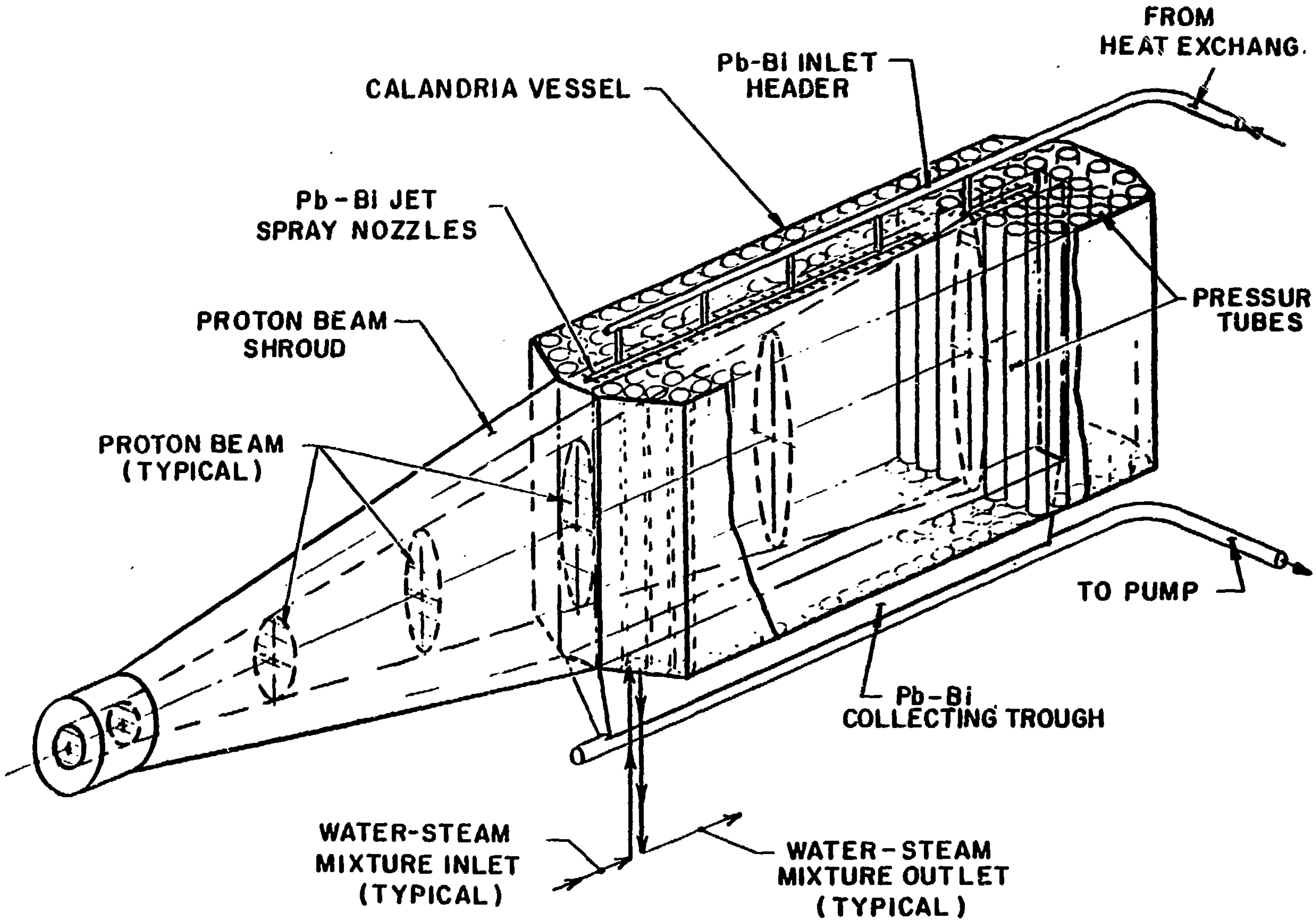
NEUTRONS PRODUCED BY REACTIONS OF GREATER THAN
OR EQUAL TO 15 MEV EXCITATION ENERGY
NEUTRONS/1 GEV PROTON

TARGET MAT.		AVER. YIELD*
PB		35
UO_2		35
1.4 H_2O/UO_2 EFFECTIVE VOL. RATIO		28
0.7 H_2O	"	29
0.17 "	"	30

INFINITE MEDIUM, 1 GEV PROTON INJECTED INTO CENTER

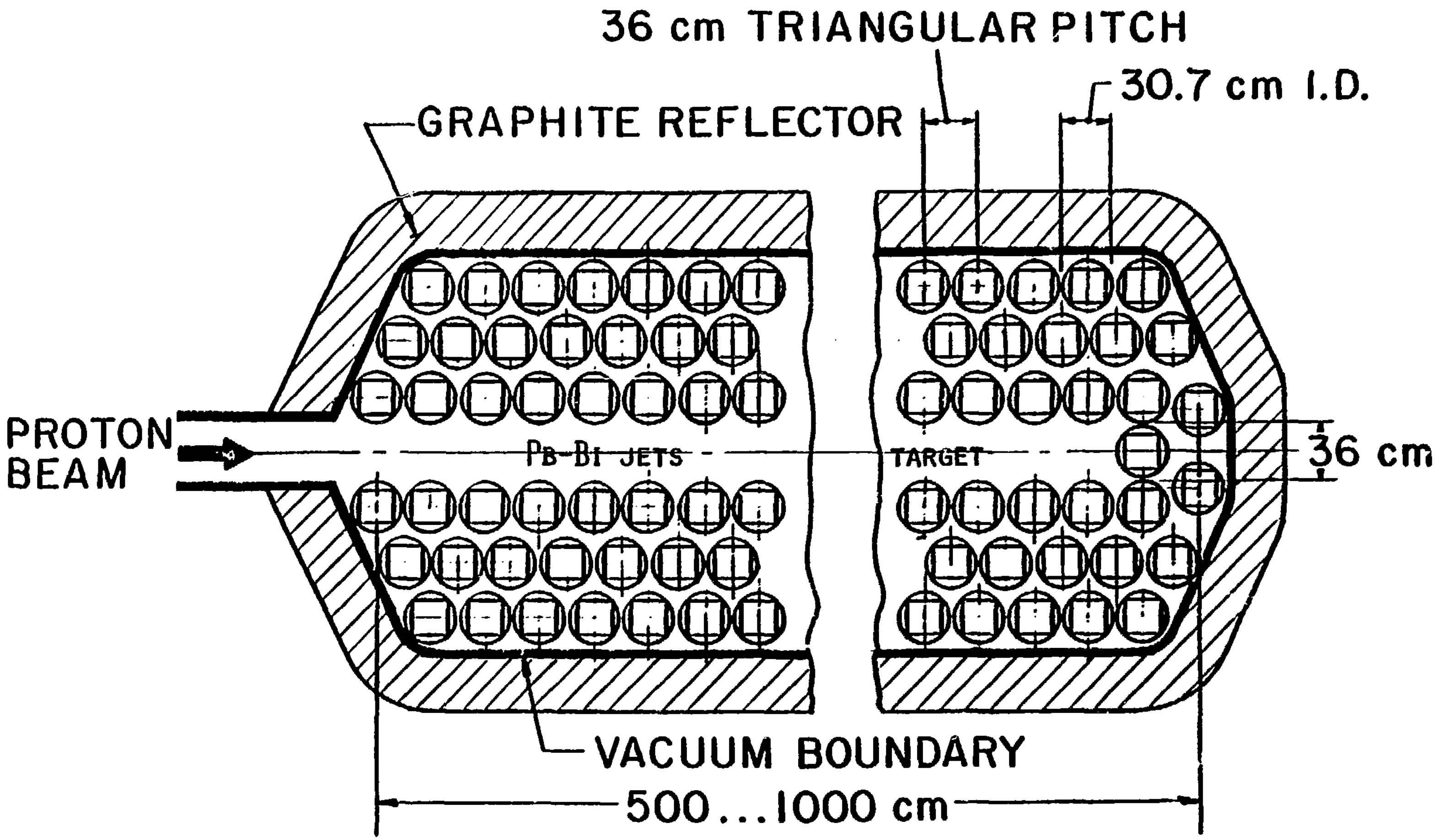
*DOES NOT INCLUDE FISSION NEUTRONS < 15 MEV.

FIGURE 4



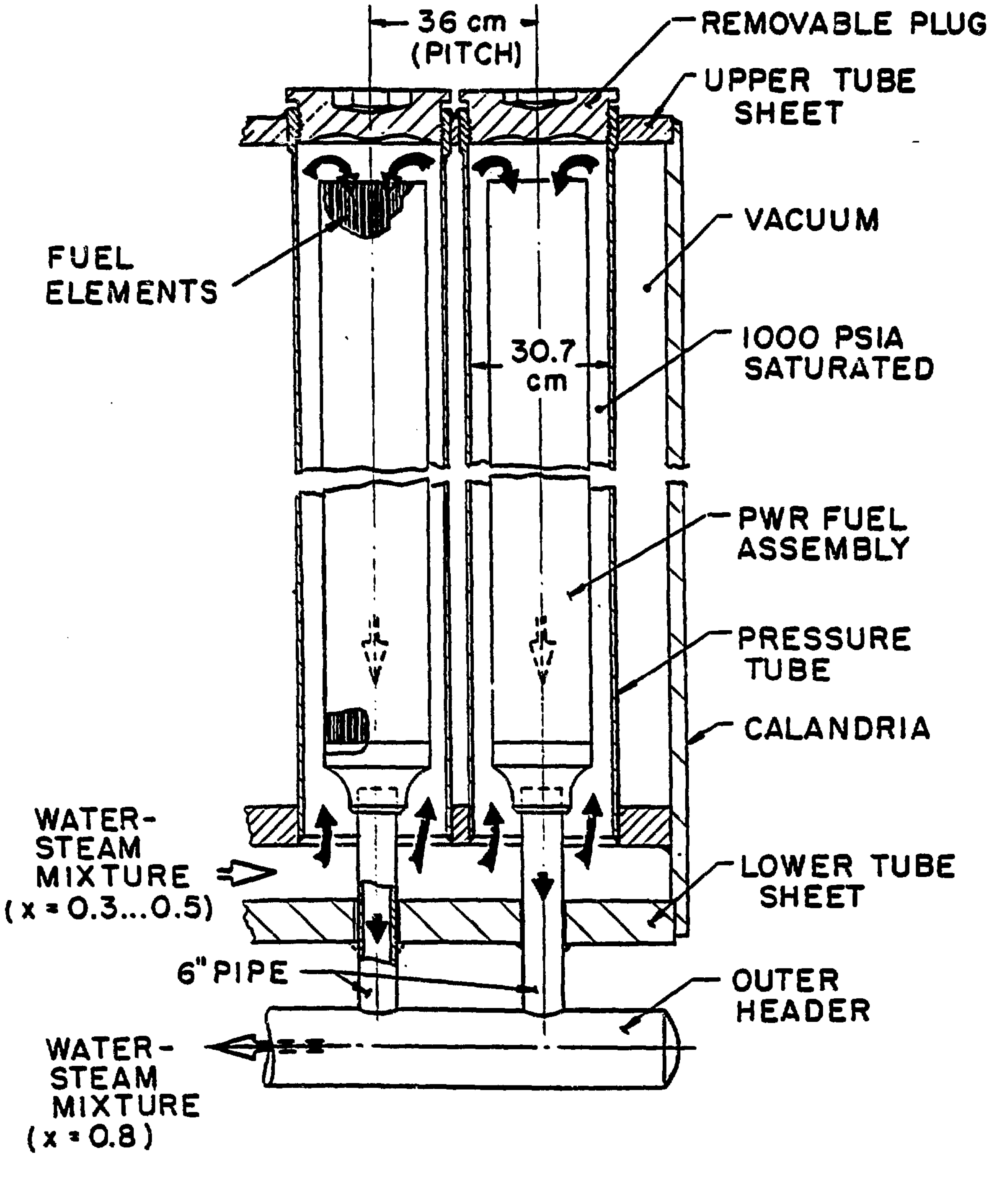
PWR-TYPE TARGET ASSEMBLY

FIGURE 5



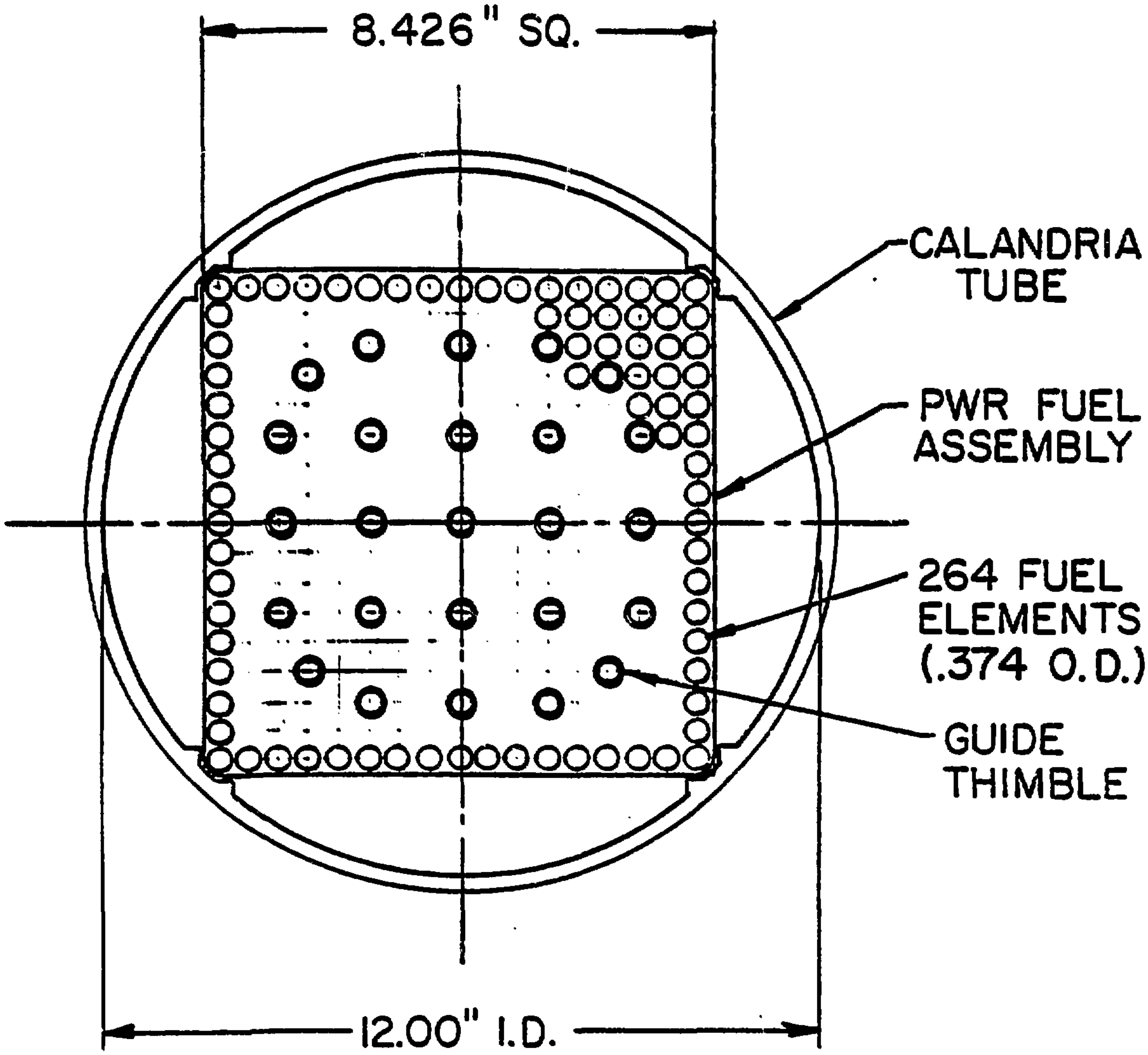
PWR TARGET ASSEMBLY
CROSS SECTION

FIGURE 6



PWR-TYPE PRESSURE TUBE ASSEMBLY

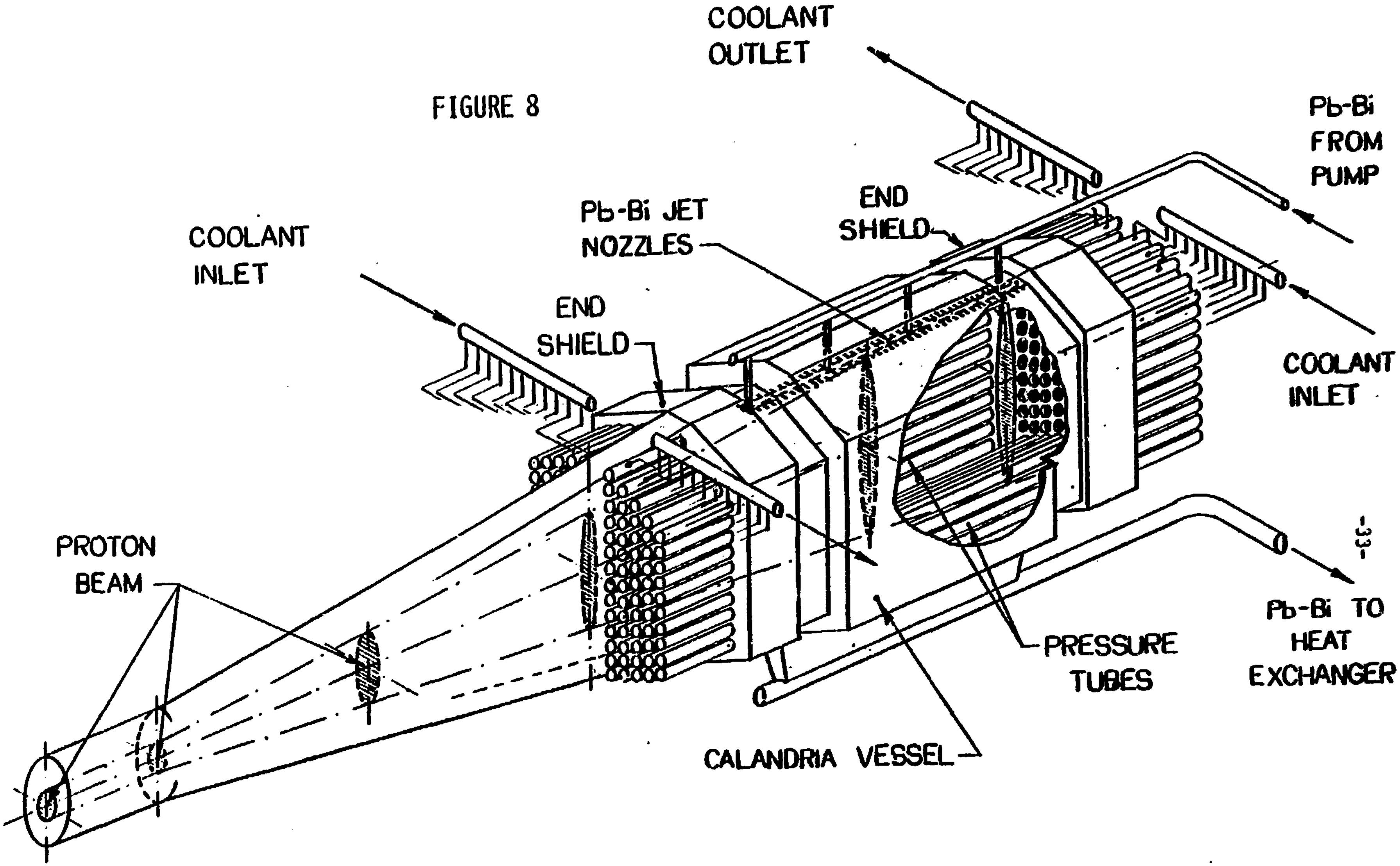
FIGURE 7



NOTE: THE GUIDE THIMBLES ACCOMODATE THE PWR CONTROL CLUSTER ELEMENT

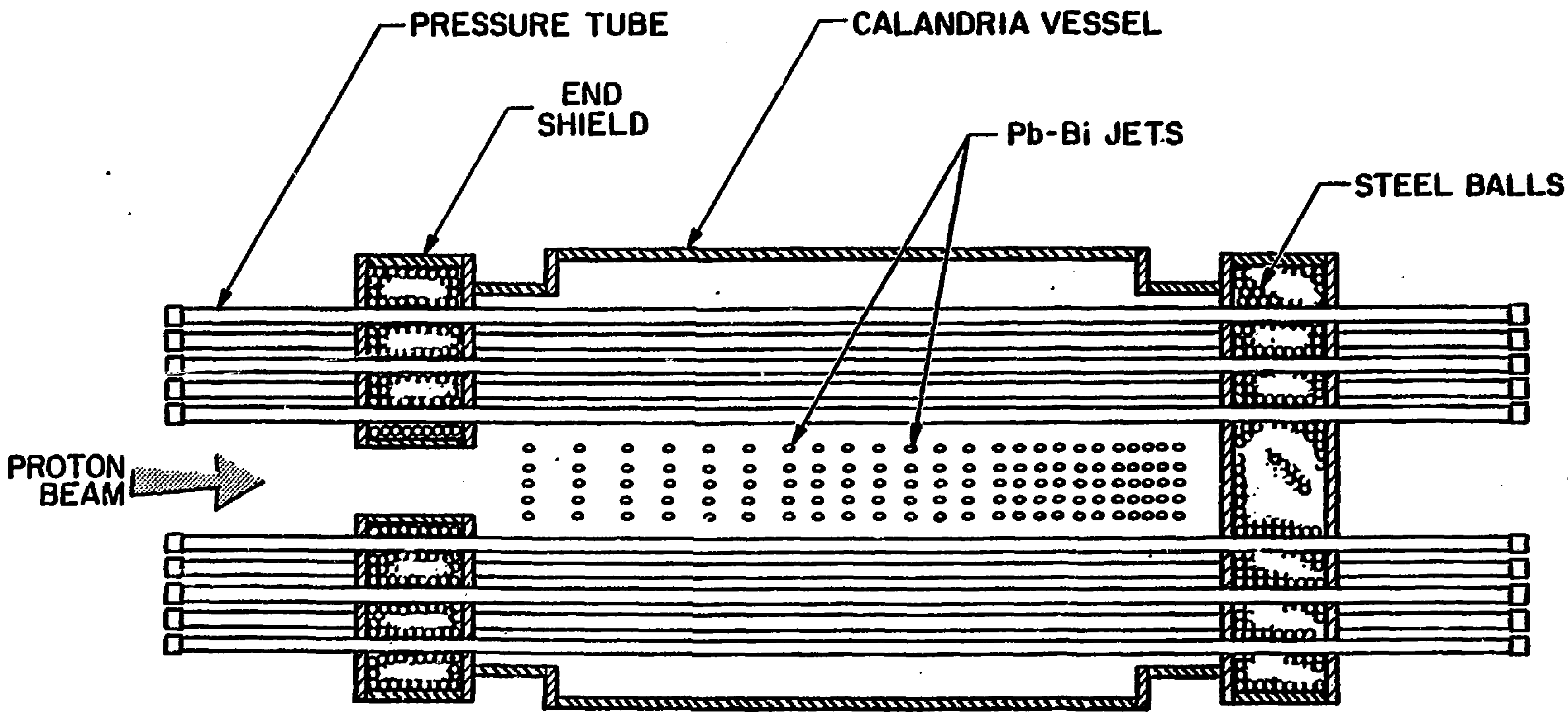
PWR PRESSURE TUBE ASSEMBLY
CROSS SECTION

FIGURE 8



HWR TYPE TARGET ASSEMBLY

FIGURE 9



HWR TYPE TARGET ASSEMBLY, CROSS SECTION

FIGURE 10

DIAGRAM OF CALCULATION PROCEDURE USED AT BNL

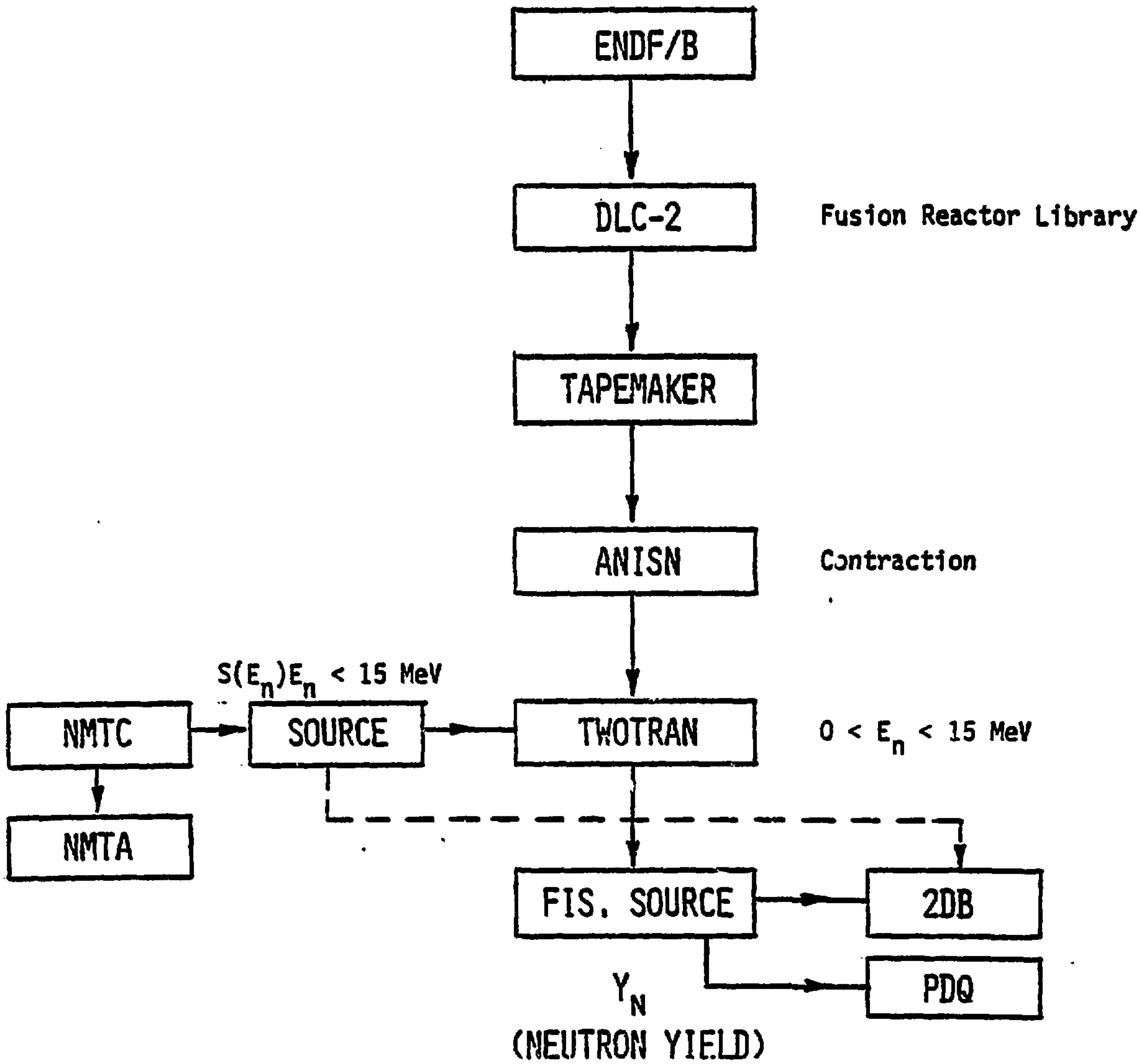


FIGURE 11

MONTE CARLO CALCULATIONS FOR Pb AND UO_2

NEUTRONS PRODUCED BY REACTIONS OF GREATER THAN OR EQUAL TO 15 MeV EXCITATION ENERGY NEUTRONS/1 GEV PROTON

BATCH		Pb	UO_2
1		37.84	35.24
2		34.32	32.04
3		36.16	34.08
4		33.60	38.80
5		37.96	36.12
6		30.36	37.52
7		30.44	34.84
8		38.76	38.68
9		34.00	29.80
10		38.48	33.68
AVERAGE Y	=	35.192	35.080 *

INFINITE MEDIUM, 1 GeV PROTON INJECTED INTO CENTER
LOWEST ENERGY 15 MeV.

*DOES NOT INCLUDE FISSION NEUTRONS < 15 MEV.

FIGURE 12

MONTE CARLO CALCULATIONS FOR WATER/UO_2 SYSTEM

NEUTRONS PRODUCED BY REACTIONS GREATER THAN OR EQUAL TO 15 MeV EXCITATION ENERGY

NEUTRONS/1 GEV PROTON

(V_{H_2O}/V_{UO_2}); (ρH_2O)

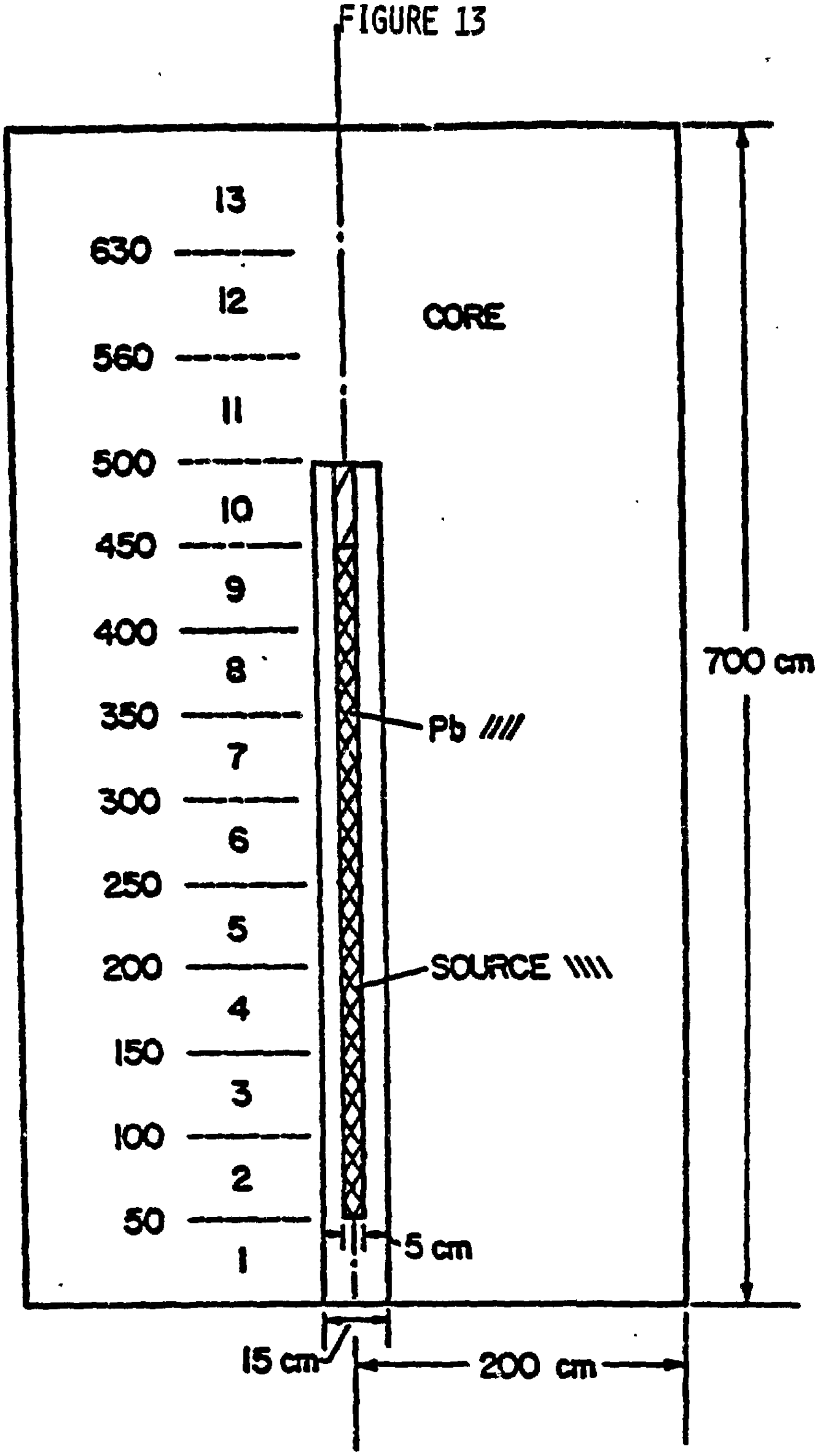
BATCH #	(2);(0.7)	(1);(0.7)	(.5);(0.7)	(.5);(0.35)	(.5);(0.175)
1	28.00	31.80	31.84	29.92	36.24
2	21.84	27.68	23.60	33.52	35.96
3	34.44	28.92	28.48	27.40	29.84
4	29.04	30.40	21.52	27.64	28.76
5	29.16	25.84	33.48	31.16	30.56
6	21.24	29.64	26.84	38.72	26.64
7	31.04	24.76	30.80	28.28	30.72
8	24.64	27.76	28.16	36.24	30.00
9	30.52	28.80	27.16	27.48	34.28
10	26.40	36.16	32.32	27.24	26.32
AVG. Y =	27.632	29.176	28.420	29.760	30.932

NATURAL URANIUM-WATER

INFINITE MEDIUM

1 GeV PROTON INJECTED INTO CENTER

FIGURE 13



GEOMETRY OF BNL-CONCEPTUAL DESIGN

FISSILE FUEL PRODUCTION CHARACTERISTICS FOR PB-BI/FUEL ELEMENT TARGET ASSEMBLIES PROTON ACCELERATOR 300 MA - 1 GEV

DESIGN NUMBER	INITIAL NEUTRON YIELD Y_N (INCLUDES FISSION REACTION)	INITIAL PRODUCTION RATE OF FUEL MATERIAL (TON/YR)	FERTILE MATERIAL	COOLANT	DENSITY OF COOLANT (G/CC)
1	35.9	Pu - 0.93	UO_2	D_2O	0.7
2	31.1	U^{233} - 0.81	TH	D_2O	0.7
3	49.4	Pu - 1.28	UO_2	H_2O	0.7
4	43.9	Pu - 1.14	UO_2	H_2O	0.35
5	43.3	Pu - 1.13	UO_2	H_2O	0.175
6	31.0	U^{233} - 0.81	TH	H_2O	0.7
7	32.4	U^{233} - 0.84	TH	H_2O	0.35
8	32.7	U^{233} - 0.85	TH	H_2O	0.175

MODERATOR/FUEL VOLUME RATIO = 0.8

FIGURE 15

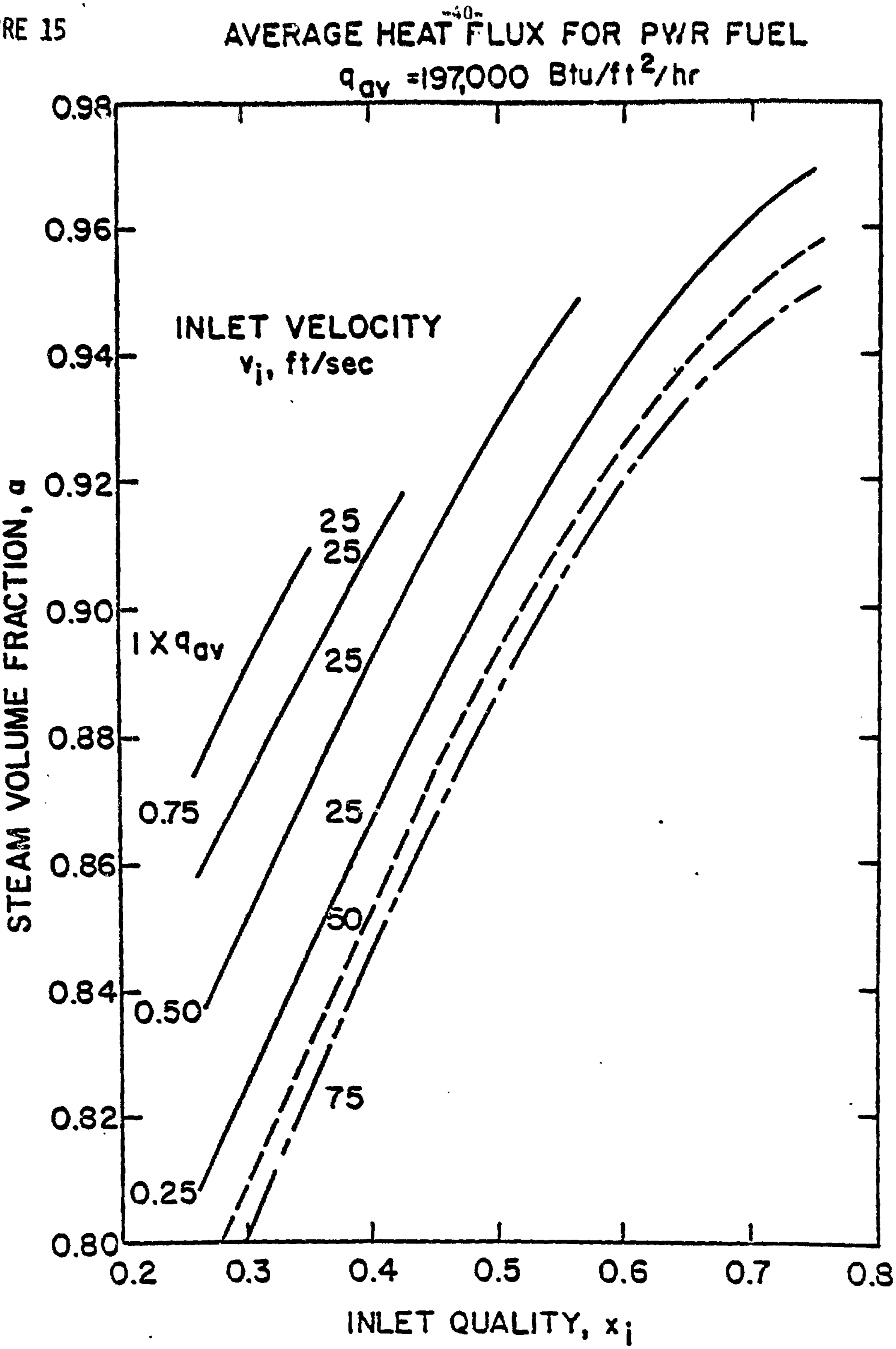

AVERAGE HEAT FLUX FOR PWR FUEL
q_{av} =197,000 Btu/ft²/hr
INLET VELOCITY
v_i, ft/sec
1 X q_{av}
0.75
0.50
0.25
25
25
25
25
50
75
0.98
0.96
0.94
0.92
0.90
0.88
0.86
0.84
0.82
0.80
0.2
0.3
0.4
0.5
0.6
0.7
0.8
STEAM VOLUME FRACTION, a
INLET QUALITY, x_i

FIGURE 16

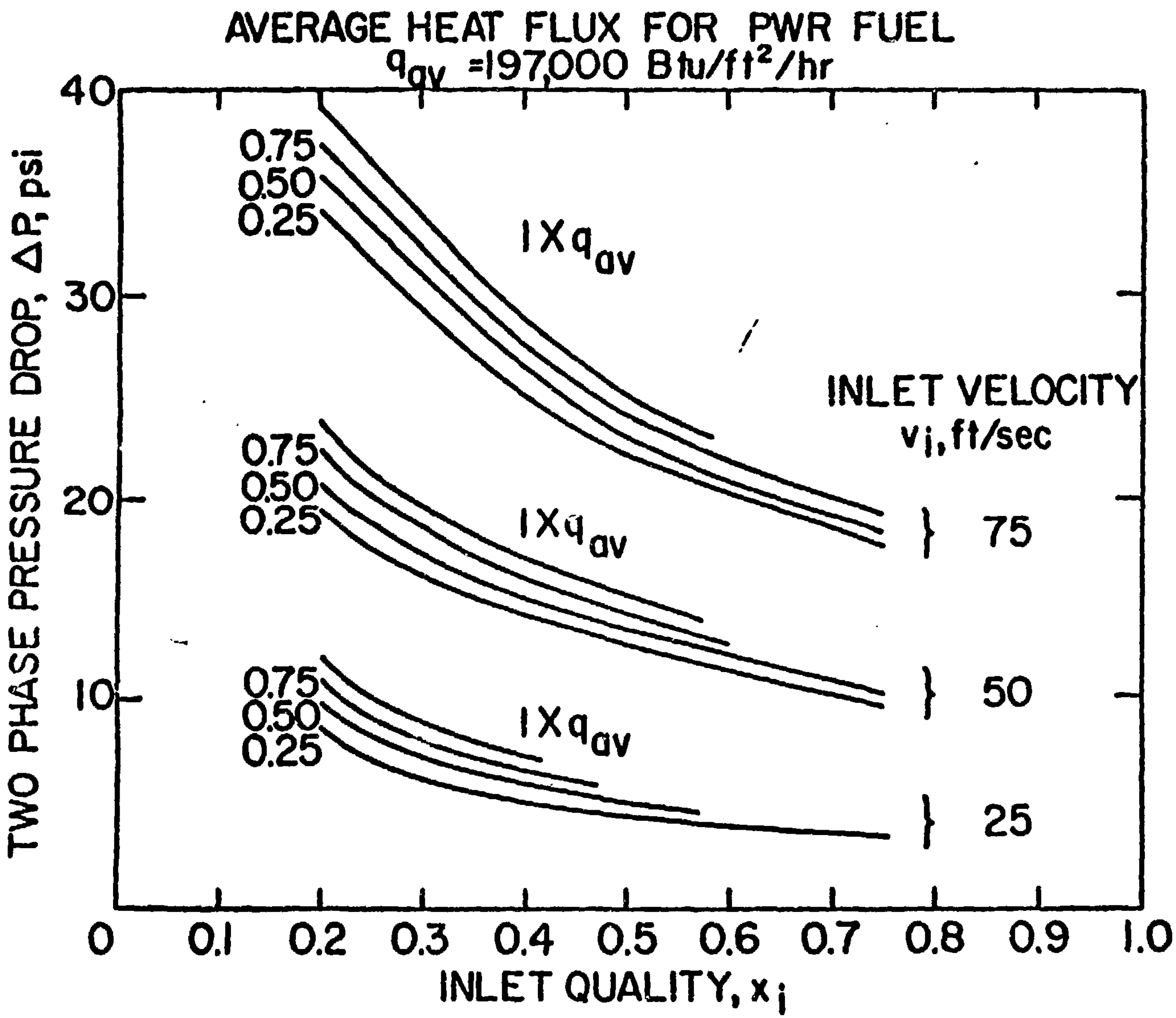
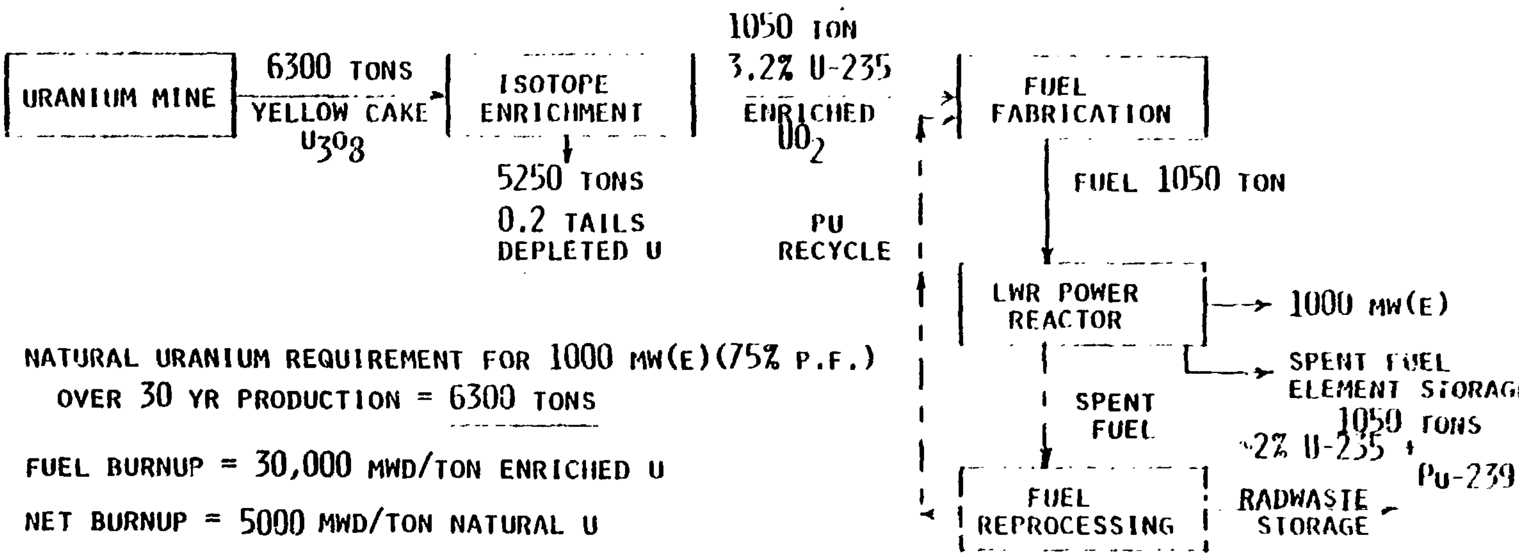

AVERAGE HEAT FLUX FOR PWR FUEL
q_{av} =197,000 Btu/ft²/hr
TWO PHASE PRESSURE DROP, ΔP, psi
40
30
20
10
0.75
0.50
0.25
1 X q_{av}
INLET VELOCITY
v_i, ft/sec
75
50
25
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
INLET QUALITY, x_i

FIGURE 17

CONVENTIONAL NUCLEAR FUEL CYCLE - LWR



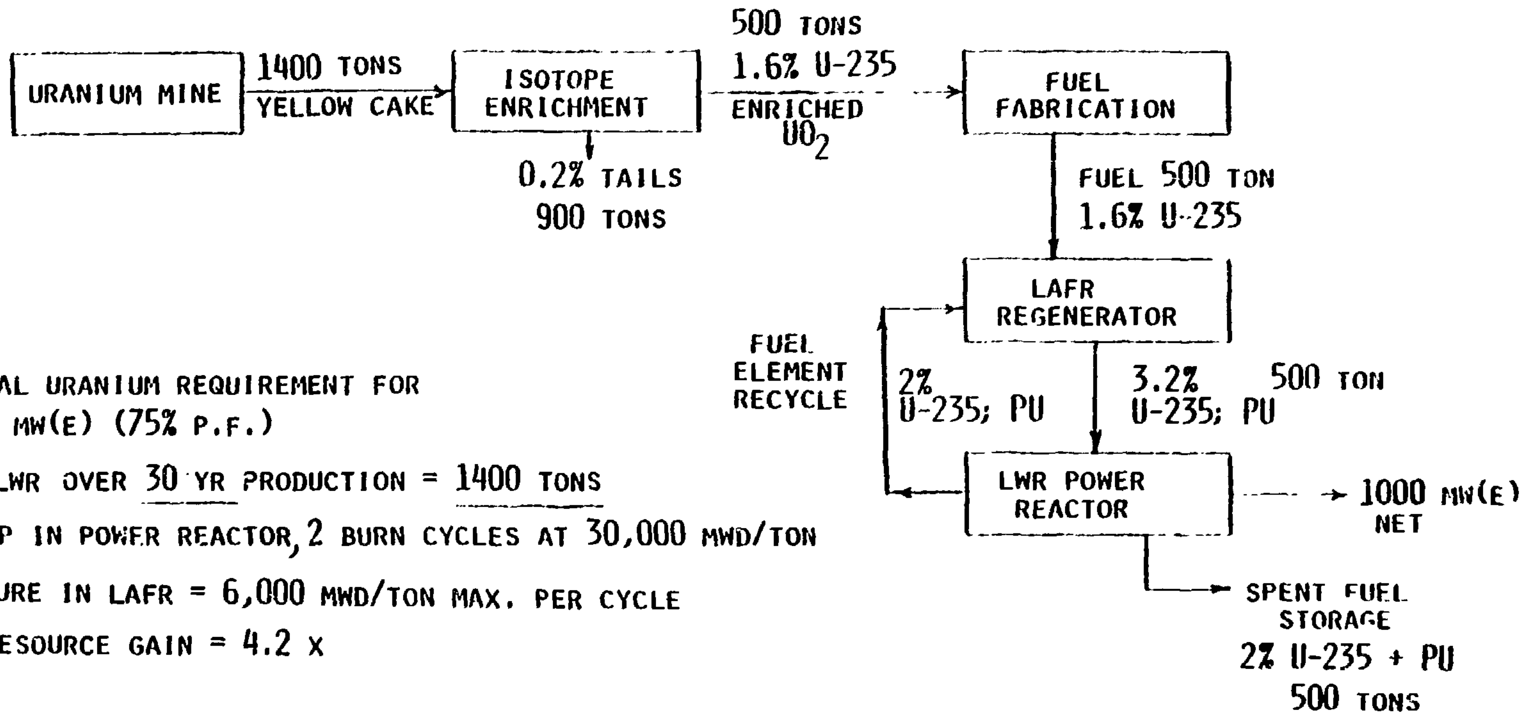
NATURAL URANIUM REQUIREMENT FOR 1000 MW(E)(75% P.F.)
OVER 30 YR PRODUCTION = 6300 TONS

FUEL BURNUP = 30,000 MWD/TON ENRICHED U

NET BURNUP = 5000 MWD/TON NATURAL U

FIGURE 13

LAFR FUEL CYCLE - NO REPROCESSING



NATURAL URANIUM REQUIREMENT FOR
1000 MW(E) (75% P.F.)

LAFR-LWR OVER 30 YR PRODUCTION = 1400 TONS

BURNUP IN POWER REACTOR, 2 BURN CYCLES AT 30,000 MWD/TON

EXPOSURE IN LAFR = 6,000 MWD/TON MAX. PER CYCLE

NET RESOURCE GAIN = 4.2 X

FIGURE 19

FUELING CYCLE FOR LAFR-LWR SYSTEM

1000 MW(E) LWR

REACTOR CORE LOADING = 100 MT

ANNUAL RELOAD = 33 MT

END OF REACTOR YEAR	LOAD IDENTITY NO. IN LAFR	LOAD IDENTITY NO. IN LWR
0	1-2-3	0
1	4	1-2-3
2	3	4-1-2
3	2	3-4-1
4	1	2-3-4
5	4	1-2-3
6	3	4-1-2
7	2	3-4-1
8	5-6-7	2-3-4
9	8	5-6-7

EACH LOAD IN LWR FOR 6 YEARS AT 10,000 MWD/MT = 60,000 MWD/MT

FIRST CYCLE IS 8 YEARS - (ON EQUIL. CYCLE 6 YEARS)

ONE LWR CORE LOADING NAT U REQUIREMENT = 2.8 x 100 = 280   ENR. FACTOR (0.7 TO 1.6%) = 2.8

+ INVENTORY IN LAFR = 33 x 2.8 = 92

TOTAL = 372 MT

- TOTAL 30 YR NAT U REQUIREMENT = 30/8 x (372) = 1400 MT

RESOURCE GAIN = 6300/1400 = 4.5 x

IN 3 REGION CORE 3.2%-2.8%-2.4% → 2.0% OUT

- ON EQUILIBRIUM BASIS 30/6 x 280 = 1400 MT

FIGURE 20

LINEAR ACCELERATOR FUEL REGENERATOR FOR LWR ECONOMY

1 LAFR FEEDING 3 LWRS

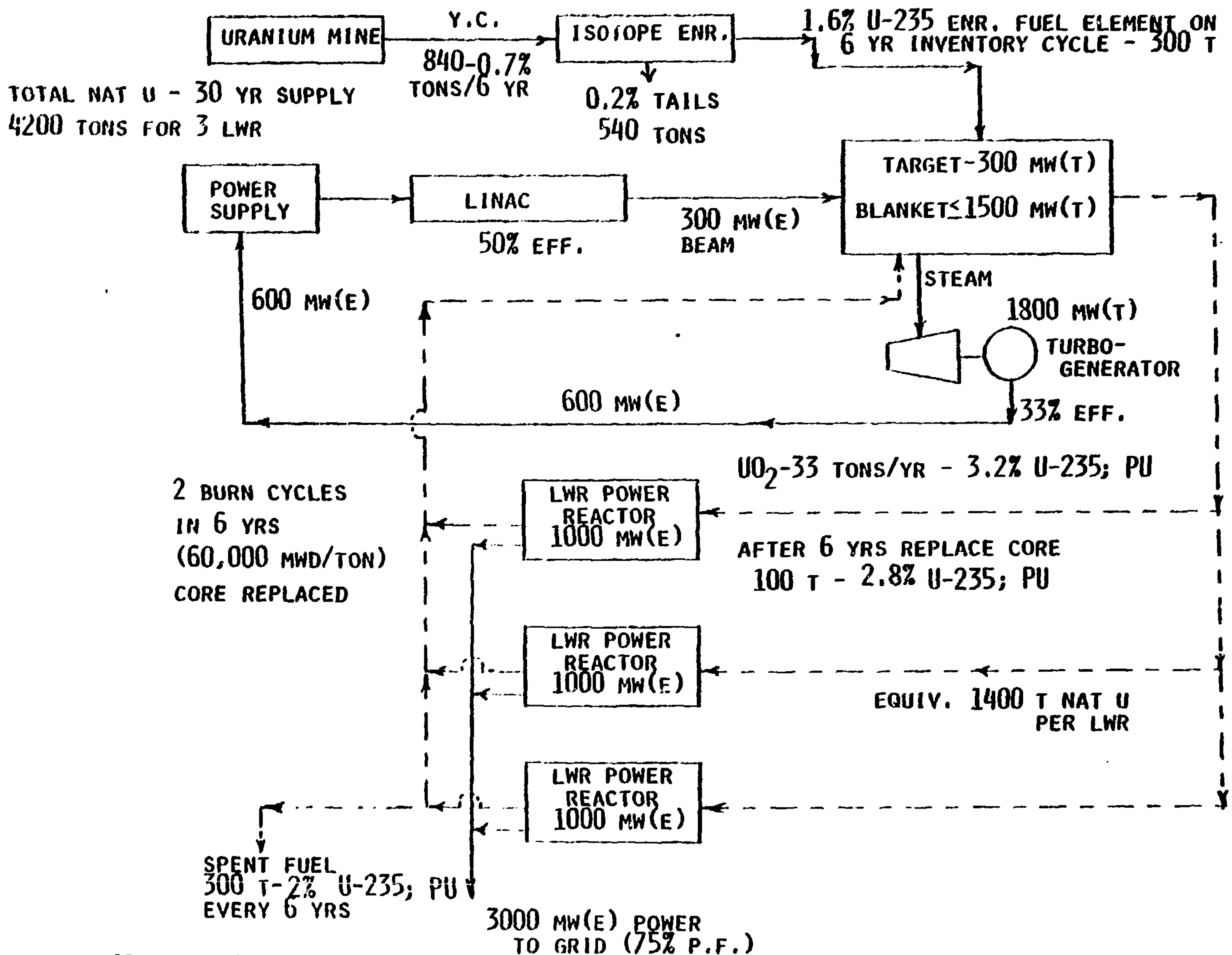
NATURAL URANIUM RESOURCE GAIN
OVER PRESENT LWR ECONOMY
30 YR LIFETIME - 1000 MW(E) PWR NEEDS - 6300 MT NAT UO_2

NO. OF BURN CYCLES AT 30,000 MWD/MT EA.	TOTAL LWR BURNUP MWD/MT	1 LAFR/3 LWR NAT UO_2 NEEDS(*) MT	NATURAL U RESOURCE GAIN LAFR-3 LWR/LWR
1	30,000	2,800	2.3
2	60,000	1,400	4.5
3	90,000	933	6.8
4	120,000	700	9.0
5	150,000	560	11.3

(*) FOR 1.6% U-235 ENRICHMENT FEED TO LAFR.

FIGURE 21

LINEAR ACCELERATOR FUEL REGENERATOR WITH LIGHT WATER REACTORS (1 LAFR/3 LWR)



TOTAL - 30 YR - 1400 MT SPENT FUEL

FIGURE 22

COMPARATIVE ECONOMICS OF LINEAR ACCELERATOR FUEL REGENERATOR WITH LIGHT WATER REACTORS

BASE CAPITAL COST FOR LAFR + 3-1000 MW(E) LWRs

(1977 DOLLARS)

LINAC 600 MW(E) x 1000 x $\frac{\$600}{KW(E)}$ = $\$360 \times 10^6$

TARGET REACTOR = 600 MW(E) x 1000 x $\frac{\$600}{KW(E)}$ = $\$360 \times 10^6$

LAFR = 720×10^6

COST OF 3-1000 MW(E) LWRS = 3 x 1000 x 1000 x $\frac{\$600}{KW(E)}$ = $1,800 \times 10^6$

TOTAL DIRECT CAPITAL COST = $2,520 \times 10^6$

UNIT COST OF LAFR PLUS 3-1000 MW(E) = $\frac{2,520 \times 10^6}{3000}$ = \$ 840/KW(E)
WITH NET POWER = 3000 MW(E)

UNIT COST OF LWR = \$ 600/KW(E)

UNIT CAPITAL COST RATIO LAFR-LWR/LWR = 1.4

FIGURE 23

COMPARATIVE ECONOMICS OF LINEAR ACCELERATOR FUEL REGENERATOR (LAFR) WITH LIGHT WATER REACTORS (LWR)

	CONV. LWR*	LAFR-LWR
CAPITAL COST		
BASE CAPITAL COST $/KW	$ 600	$ 840
COMPLETION COST $/KW (ESCALATED FOR 1986 OPERATION)	1100	1540
RESOURCE		
FUEL REQUIREMENT OVER 30 YEARS, TONS NAT. U	6300	1400
POWER GENERATION COST (AVERAGE 1ST-10 YRS)	MILLS/KWH(E)	
CAPITAL CHARGES (15% AND 70% L.F.)	26.9	37.7
FUEL (5% ESCALATION/YR)	13.3	4.4
OP. & MAINT.	3.3	4.0
TOTAL	43.5*	46.1

(*) ANS INDUSTRY REPORT (1976-7) - BECHTEL ESTIMATE.

CONCLUSION - LAFR/3 LWR REASONABLY COMPETITIVE WITH LWR.

FIGURE 24

FUEL CYCLE COST - MILLS/KWH(E)

(ESCALATED TO 1986)

	UNIT COST 1977 DOLLARS	LWR	REDUCTION FACTOR	2-CYCLE LAFR-LWR
YELLOW CAKE	\$50/LB U_3O_8	3.94	4.5	0.88
CONVERSION	\$11/KG	0.29	4.5	0.06
ENRICHMENT	\$100/SWU	2.90	9.0	0.32
FABRICATION	\$200/KG	2.05	2.0	1.03
STORAGE AND CARRYING CHARGE	\$400/KG HM	3.87	2.0	1.94
TRANSPORTATION	\$30/KG HM	0.31	2.0	0.15
		13.36*		4.38

(*) INDUSTRY REPORT (1976-7) - BECHTEL ESTIMATE.

FIGURE 25

FISSILE FUEL PRODUCTION ECONOMICS

CAPACITY	LAFR OR LAFP
BEAM POWER - MW(E)	300
POWER TO ACCEL. - MW(E)	600
RECIRCULATING POWER - MW(E) (MAX)	600
OUTSIDE POWER REQUIRED - MW(E)	0
FUEL PRODUCTION RATE - KG/YR	1200 Pu-239 OR 1180 U-233
CAPITAL COST	
LINAC - 300 MW(E) BEAM	$\$360 \times 10^6$
TARGET 600 MW(E)	360×10^6
DIRECT CAPITAL	720×10^6
COMPLETED COST (ESCALATED 1986)	$1{,}320 \times 10^6$

PRODUCTION COST OF U-233 - FISSILE MATERIAL	\$/GM
DEPRECIATION (15% PER ANNUM)	165.00
FUEL CYCLE	80.00
O&M	12.00
PRODUCTION COST - \$/GM FISSILE MAT. PRODUCED & CONSUMED	\$ 257.00

FABRICATED LEU FUEL CYCLE COST = 13.3 MILLS/KWH(E) X $\dfrac{\text{18,200 KWH(E)}}{\text{GM FISSILE CONS.}}$ = \$242.00/GM NET FISSILE MAT. CONSUMED
(CONV. RATIO = 0.6)

(\$96.00/GM INITIAL U-235 CONTAINED IN FUEL ELEMENT)

CONCLUSION - LAFR FUEL COST WITHIN 10% OF U-235 COST.

FIGURE 26

CONCLUSIONS FROM ECONOMIC ESTIMATES

- LAFR IN RANGE OF LWR COST - TRADEOFF FUEL COST FOR CAPITAL COST
- EXTEND FUEL SUPPLY 4.5 TIMES BASED ON
  - A) BURNUP OF 60,000 MWD/TON
  - B) NO REPROCESSING
  - C) THROWAWAY FUEL CYCLE
- REDUCES ENRICHMENT PLANT REQUIREMENTS SIGNIFICANTLY (EST. 9.0 X LESS SWU'S)
- USES LWR POWER TECHNOLOGY
- LAFR IS AN INDEPENDENT SOURCE OF SUPPLY OF IN-SITU ENRICHED FUEL
- LAFR LOOKS EVEN BETTER WITH THORIUM FUEL CYCLE (LWR C.R. $\geq$ 0.73)
  (HWR C.R. $\geq$ 0.90)
- COST DECREASES LESS FOR GREATER THAN 2 BURN CYCLES
- INCREASES RESOURCE UTILIZATION SIGNIFICANTLY

FIGURE 27

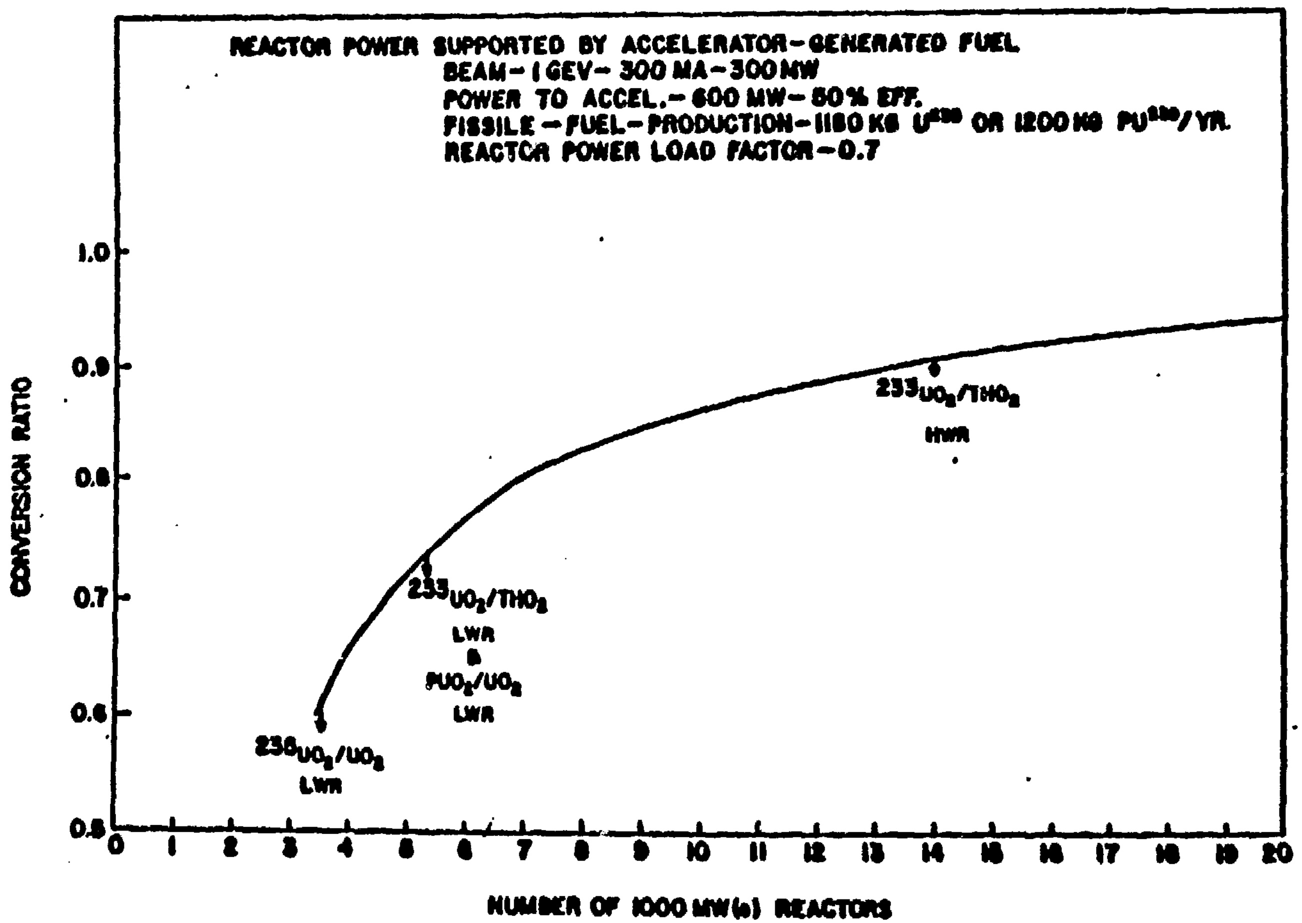

REACTOR POWER SUPPORTED BY ACCELERATOR-GENERATED FUEL
BEAM-1 GEV-300 MA-300 MW
POWER TO ACCEL.-600 MW-50% EFF.
FISSILE-FUEL-PRODUCTION-1180 KG U OR 1200 KG PU/YR.
REACTOR POWER LOAD FACTOR-0.7
CONVERSION RATIO
1.0
0.9
0.8
0.7
0.6
0.5
235UO2/UO2
LWR
233UO2/THO2
LWR
&
PUO2/UO2
LWR
233UO2/THO2
HWR
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
NUMBER OF 1000 MW(e) REACTORS

FIGURE 28

REACTOR POWER SUPPORTED BY ACCELERATOR GENERATED FUEL
POWER COST FOR U AND TH FUEL CYCLES

ACCEL. BEAM POWER = 300 MW - POWER TO ACCEL. = 600 MW(E)
FISSILE FUEL PRODUCTION - 1180 KG U^{233} OR 1200 KG PU^{239}/YR
REACTOR POWER LOAD FACTOR = 0.7

REACTOR TYPE	LWR	LWR	HWR
FUEL FROM LAFP	$^{239}PU/^{235}UO_2/UO_2$	$^{233}UO_2/THO_2$	$^{233}UO_2/THO_2$
CONVERSION RATIO	0.6	0.73	0.9
TOTAL REACTOR POWER SUPPORTED, MW(E)	3400	5000	13,600
POWER COST FOR 1000 MW(E) REACTOR		MILLS/KWH(E)	
FUEL CYCLE	14.1	9.5	3.5
O & M	3.3	3.3	3.3
HEAVY WATER	--	--	2.2
DEPRECIATION (15%/YR)	26.9	26.9	26.9
TOTAL PRODUCTION COST	44.3	39.7	35.9
RESOURCE ENTENDED WITH LAFR - NO REPROCESSING (X PRESENT LWR THROWAWAY)	4.5 x	6.7 x	18.0 x

LAFP: WITH REPROCESSING, RESOURCE IS EXTENDED 200 TIMES LWR THROWAWAY ECONOMY

CONCLUSION - CONVERTING TO TH AND HWR IMPROVES ECONOMICS AND RESOURCES.

FIGURE 29

LINEAR ACCELERATOR FUEL PRODUCER AND REGENERATOR

TH/U-233 TOPPING OF ENRICHED U/U-235

DENATURED FUEL CYCLE

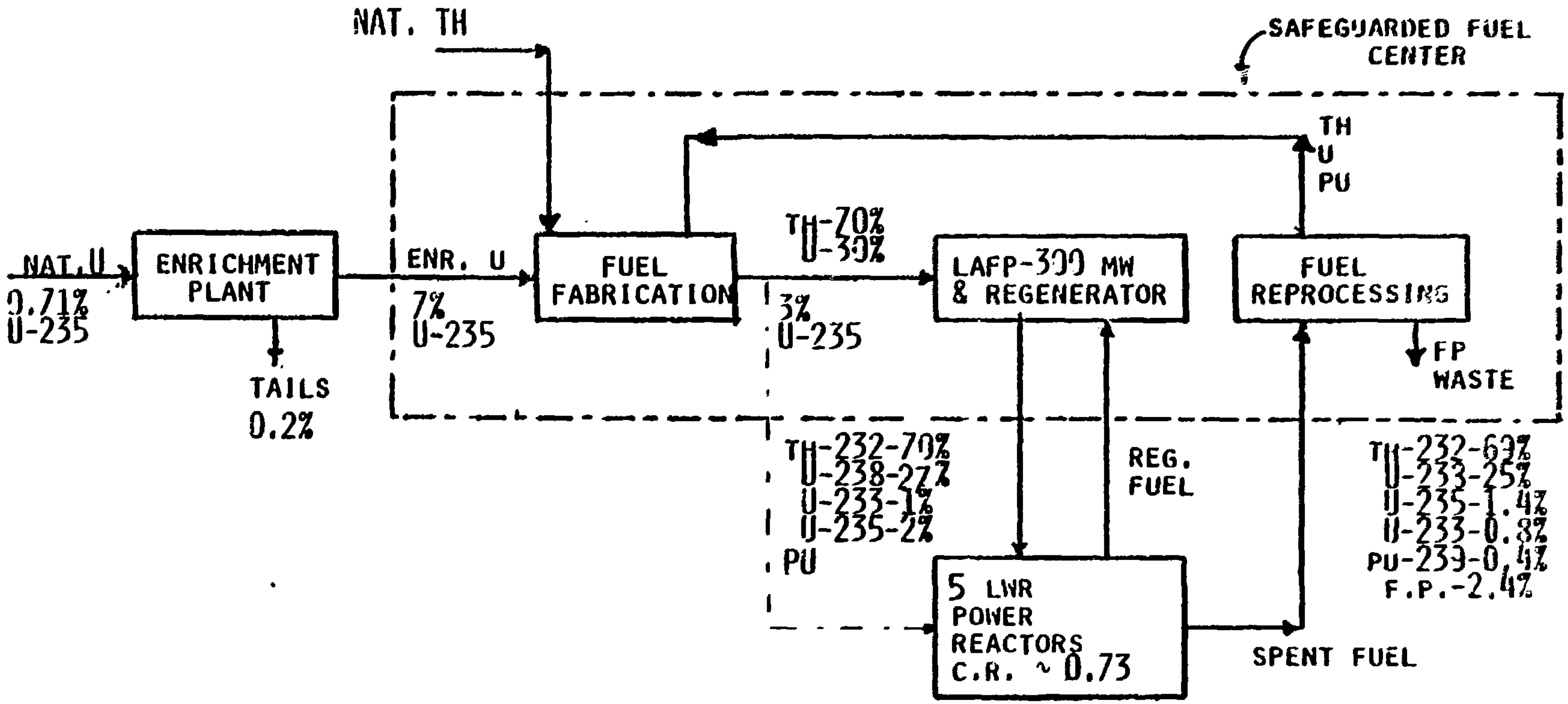


FIGURE 30

NUCLEAR POWER GROWTH PATTERNS

EFFECT OF DENATURED SYSTEMS SMALL U_3O_8 SUPPLY

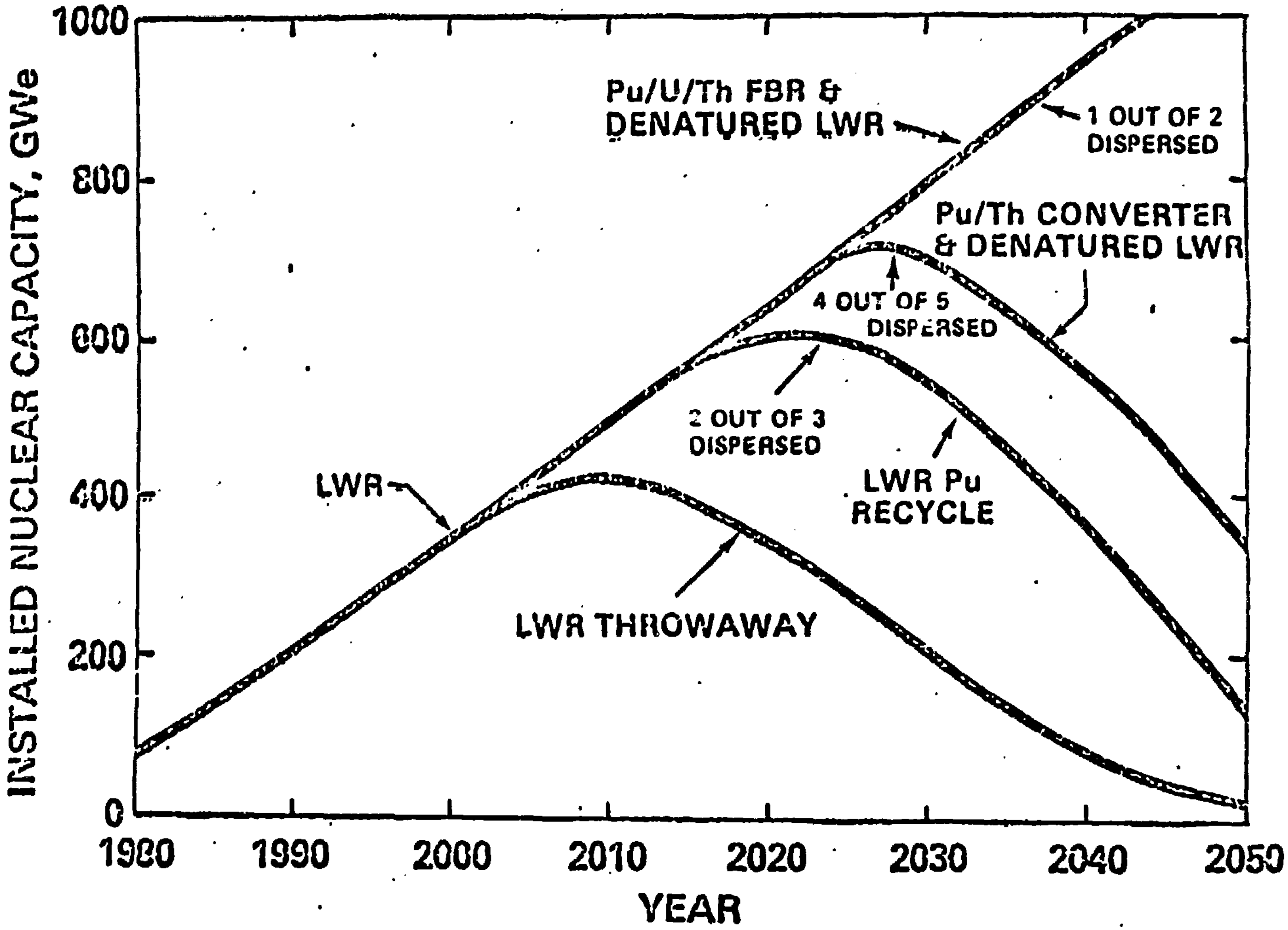


FIGURE 31

NUCLEAR POWER GROWTH PATTERNS

EFFECT OF LAFP SYSTEMS (SMALL U_3O_8 SUPPLY)

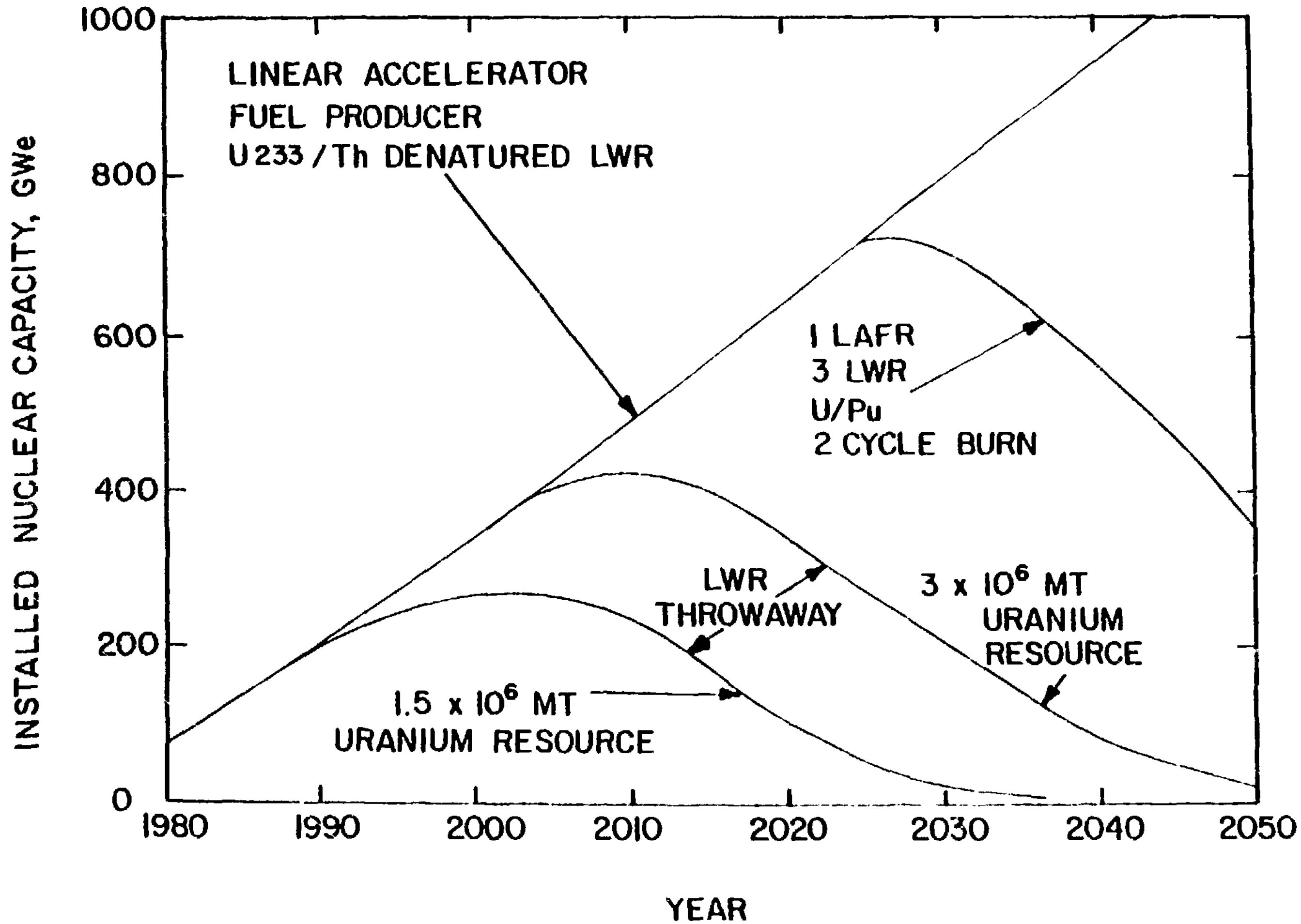


FIGURE 32 KEY PARAMETERS FOR ECONOMIC COMPARISON OF FISSILE FUEL PRODUCERS

		INTERNAL NEUTRON SOURCE		EXTERNAL NEUTRON SOURCE	
	U-235 ENRICHMENT PLANT	LWR	LMFBR	HYBRID	ACCELERATOR LAFR/LAFP
CAPITAL COST (1986)	$1 x 10^9 1.5 x 10^6 SWU	$1 x 10^9	$1.4 x 10^9	$3.7 x 10^9	$1.3 x 10^9
PU PRODUCTION MT/YR	10 (U-235)	0.3	0.1-0.3	1.5	1.0
OR U-233 PRODUCTION MT/YR	--	0.4	0.1-0.3	1.5	1.0
POWER DELIVERED MW(E)	-65 MW CONS	1000	1000	500	0
FUEL REQUIREMENT	--	SMALL FISSILE LEU OR PU SOURCE	LARGE FISSILE SOURCE	FERTILE FUEL ONLY	FERTILE FUEL ONLY
LIFETIME FERTILE FUEL REQUIREMENT, MT (NO REPROCESSING)-30 YR	60,000	6000 WITH ENRICHMENT	CANNOT FUNCTION	2100	1400 (ON TH-U CYCLE ≤500
LIFETIME FERTILE FUEL REQUIREMENT, MT (WITH REPROCESSING)-30 YR	40,000	4000	30	30	30
SYSTEM SUPPORT	1 ENRICH. PLANT SUPPORTS 10-15/LWR	1 LWR CANNOT SUPPORT ANY LWR	1 PU BREEDER SUPPORTS 1 PU/LWR NO DENATURED LWR	1 PU HYBRID SUPPORTS 5 LWR OR 20 ON TH-U	1 PU LAFP SUPPORTS 3 LWR OR 12 ON TH-
FLEXIBILITY FOR LWR ECONOMY	SEVERELY LIMITED BY NAT U RESOURCE	SEVERELY LIMITED BY NAT U RESOURCE	LIMITED BY POWER CONSUMPTION GROWTH RATE	NO LIMITATIONS	NO LIMITATIONS

FIGURE 33

COMPARISON OF LONG-TERM FISSILE FUEL PRODUCTION SYSTEM

(BASIS: 500 GW(E) NET POWER)

	FBR + LWR	LAFR + LWR U/PU	LAFR + LWR TH/U	HYBRID + LWR U/PU	HYBRID + LWR TH/U
TOTAL SYSTEM CAPITAL (1986 ESC, $) INVESTMENT ($BILLIONS)	600	725	630	780	670
NTI - NEW TECHNOLOGY INVESTMENT ($BILLIONS)	350	225	130	330	200
NO. OF LWR'S	250	500	500	450	470
SUPPLY RATIO	LWR/FBR 1/1	LWR/LAFR 3/1	5/1	LWR/HYBRID 5/1	8/1
NO. OF NTI PLANTS	250	170	100	90	55
LWR UNIT CAPITAL COST, $KW(E)	1000	1000		1000	
NTI CAPITAL COST PER PRODUCTION PLANT ($BILLION)	1.4	1.3		3.7	
R&D COST (10^9)	3	3		15	
COMMERCIAL INTRODUCTION DATE	1990	1995		> 2000 ?	

CONCLUSION FOR LAFR - TOTAL SYSTEM INVESTMENT 5-20% MORE FOR LAFR COMPARED TO FBR BUT NEW TECHNOLOGY CAPITAL INVESTMENT IS MUCH LESS (1.6 TO 2.7 TIMES LESS THAN FBR);

- AND FOR LONG TERM, THE UTILITY USES CONVENTIONAL LWR TECHNOLOGY.

FIGURE 34

UNIQUE FEATURES OF
LINEAR ACCELERATOR FUEL PRODUCER AND REGENERATOR

A LONG-TERM INDEPENDENT FUEL SUPPLIER FOR THE LWR POWER ECONOMY

- PROVIDES FLEXIBILITY IN LWR NUCLEAR FUEL CYCLE
- INSURES LONG-TERM USE OF LWR THERMAL REACTOR ECONOMY AT A COMPETITIVE COST
- OPENS UP LONG-TERM USE OF EXISTING NUCLEAR FUEL RESOURCE
- OPENS UP TH/U-233 FUEL CYCLE
- MAKES EFFICIENT USE OF EXISTING ENRICHMENT PLANTS
- DEPENDS ONLY ON EXTENSION OF EXISTING TECHNOLOGY - ACCELERATOR AND TARGET
- R&D COST OF DEMONSTRATION AND IMPLEMENTATION IS CONSIDERABLY LESS THAN FOR HYBRID
- PRODUCES A HIGHLY PROLIFERATION RESISTANT FUEL CYCLE - TH/U-233 DENATURED FUEL
- CAN LEAD TO USE OF ACCELELATOR FOR RADWASTE DISPOSAL BY TRANSMUTING LONG-LIVED FISSION PRODUCTS AND TRANSURANICS TO SHORT-LIVED AND STABLE PRODUCTS

FIGURE 35

R&D REQUIRED AND SCHEDULE FOR IMPLEMENTATION

(1978 DOLLARS)

	M$	YEAR
• TARGET PHYSICS EXPERIMENT	50	1980-1983
• ACCELERATOR AND TARGET COMPONENTS DEVEL.	500	1980-1990
• DEMO PILOT PLANT AND R&D	1,000	1990
• PROTOTYPE COMMERCIAL PLANT AND R&D	2,000	2000